September 1955

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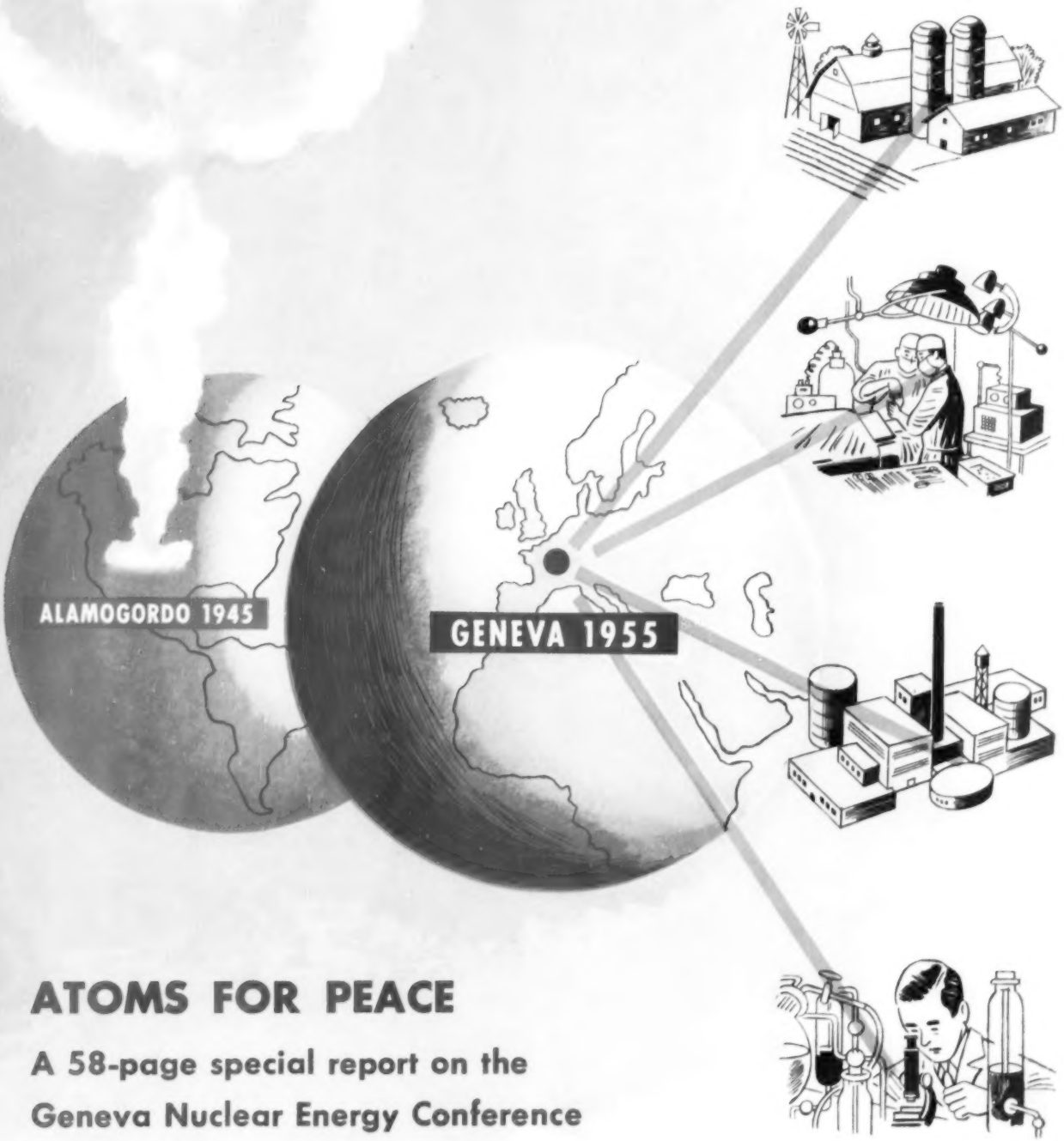

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B&W NUCLEAR STEAM

Selected for

CON EDISON'S ATOMIC POWER STATION

Progress toward large-scale use of atomic energy took a giant forward step with Consolidated Edison Company's decision to build a nuclear steam-electric generating plant at Indian Point, N.Y. The first privately financed atomic plant in history, Con Edison's 250,000 kw station will rank with the largest conventional fuel plants now being built.

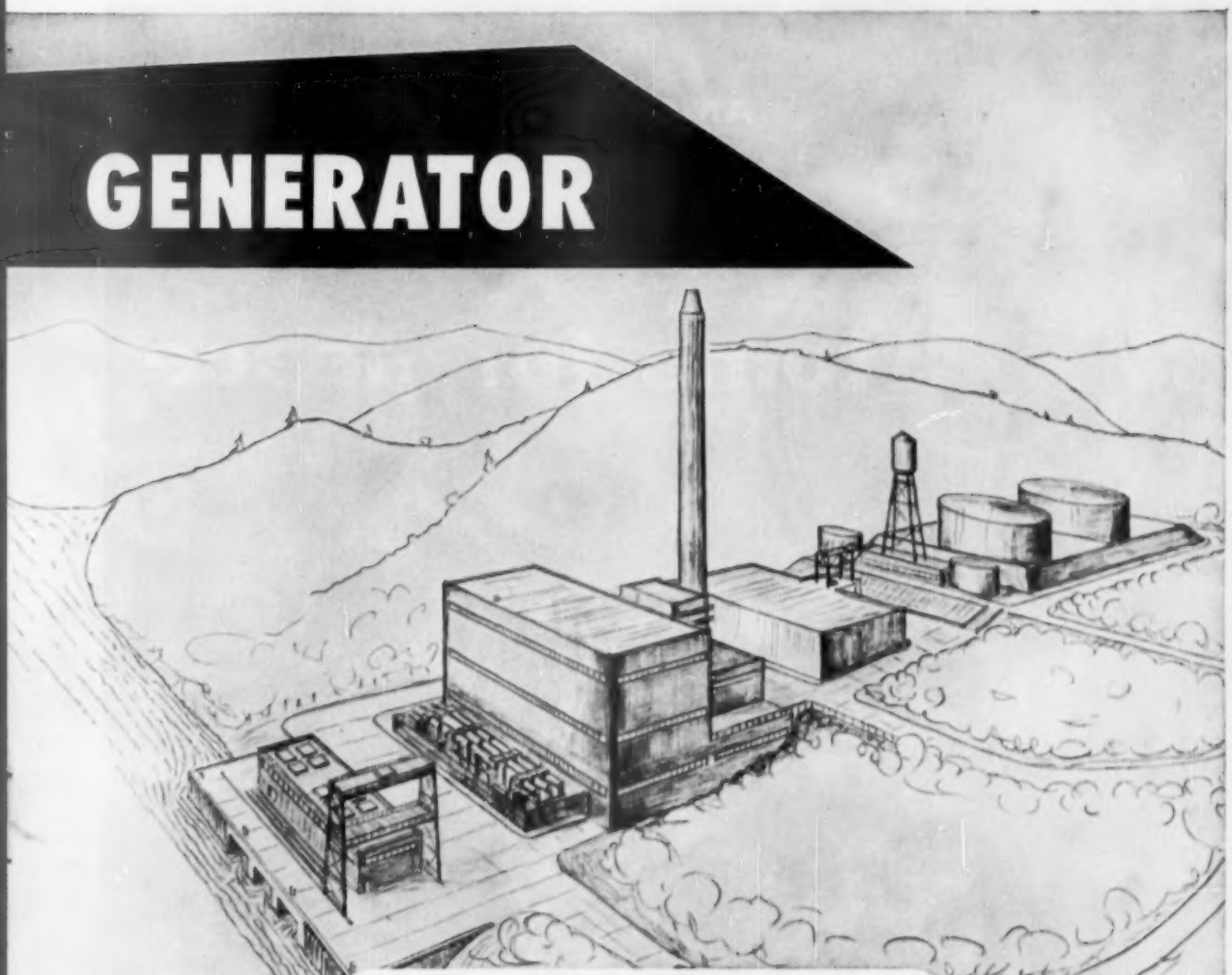
Con Edison's farsightedness in going ahead with private development of atomic energy is in the tradition established by its founders when electricity was first brought to New York City by Thomas A. Edison 73 years ago. To Con Edison, as to B&W, atomic fission represents a new and valuable source of heat — to be harnessed to develop electricity in a safe, efficient manner for the benefit of its customers.

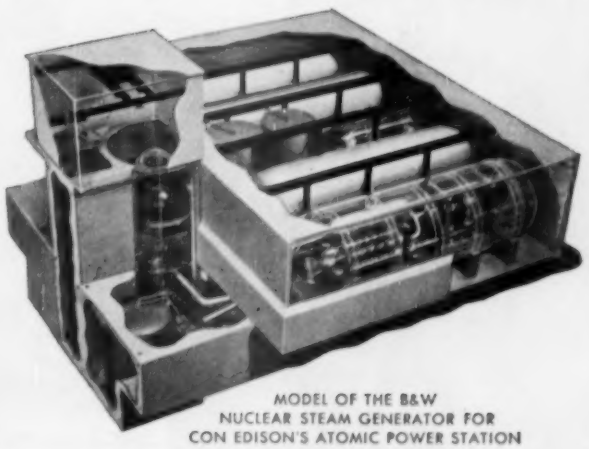
In selecting Babcock & Wilcox — an experienced supplier of steam generating equipment — to design, fabricate and erect the nuclear steam generator for the Indian Point plant, Con Edison recognized that a large share of the skills involved in nuclear power concern heat transfer, steam generation, fluid flow, high pressure piping, metallurgy, welding, and similar fields — in all of which B&W is a recognized leader.

Another basic consideration in this decision was that B&W, with its thousands of man-months of experience in nuclear power, is in a position to negotiate a fixed price contract for the required equipment, in the same manner as for a steam generator using conventional fuel. When its new fuel element fabrication plant now under construction near Lynchburg, Va., goes into operation, B&W will be capable of fabricating almost 90% of this equipment in its own shops.

Since the first days of the Manhattan Project B&W's experts have been called on for solutions to many of the special problems which arise in atomic energy. Today, more than 100 experienced B&W nuclear physicists and engineers, working in concert with B&W's large staff in conventional steam generation and its allied arts, are directing the wide range of atomic energy projects which B&W has been awarded. The best guarantee of realizing the full potentials of nuclear power at the earliest possible date is this combination of an intimate knowledge of atomic energy with nearly a century of steam generating experience. The Babcock & Wilcox Company, Atomic Energy Division, 161 East 42nd Street, New York 17, N. Y.

GENERATOR



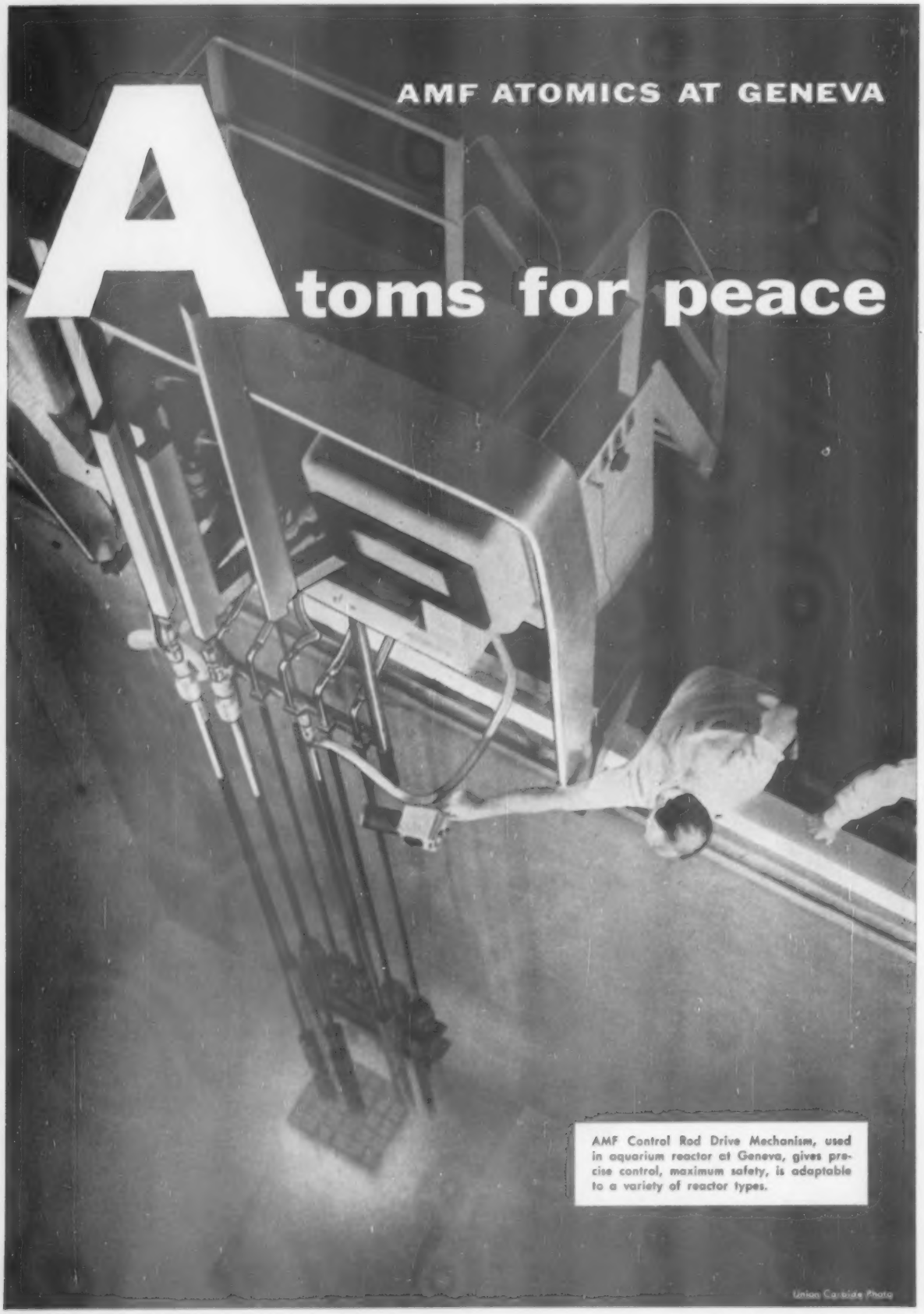


MODEL OF THE B&W
NUCLEAR STEAM GENERATOR FOR
CON EDISON'S ATOMIC POWER STATION

BABCOCK & WILCOX
ATOMIC ENERGY DIVISION

AMF ATOMICS AT GENEVA

Atoms for peace



AMF Control Rod Drive Mechanism, used in aquarium reactor at Geneva, gives precise control, maximum safety, is adaptable to a variety of reactor types.

Union Carbide Photo

. . . in action

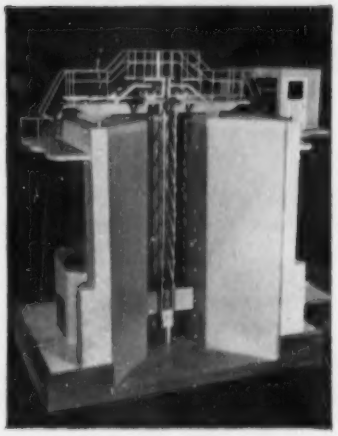
Working model of AMF Control Rod Drive Mechanism illustrates unit's flexibility, precision, and safety.

One of the largest U. S. exhibitors at the International Conference on the Peaceful Uses of Atomic Energy, AMF is in the forefront of the atomic energy field . . . translating principles into design, design into *action.*

AMF "Unitized" Control Rod Drives, for example, operated the aquarium reactor at the United States Exhibit at Geneva.

AMF-produced Master-Slave Manipulators,* already widely used for "hot labs" in this country, were on display in several exhibits, illustrating the precision handling of radioactive materials with complete safety.

A large scale model of the AMF "swimming pool" type research reactor now under construction for Battelle Memorial Institute demonstrated the "Unitized" engineering concept AMF has applied to a variety of reactors of practically every type. Featuring interchangeability of component units, this AMF "Unitized" design simplifies maintenance, allows easy modification to avoid obsolescence.



Large scale model of AMF "swimming pool" type reactor now being built demonstrates AMF's "Unitized" design as applied to reactors.

And rounding out the AMF Atomics Inc. exhibit was an actual industrial application of atomic energy – the "Nucleonic Microfeed" that makes possible absolute uniformity of weight and density in the production of cigarettes.

AMF looks forward to the next atomic energy conference, when science will inevitably have progressed much further along the road to atoms for peace. In the meantime, AMF Atomics invites your inquiry on matters nucleonic . . . on reactor design and construction, component mechanisms, materials handling, or any phase of the practical application of atomic energy.

*Modification of the Model #8 as developed by the Argonne National Laboratory

AMF-produced "Master-Slave" Manipulator for "hot cells" practically duplicates human wrist motion, makes possible delicate operations, handles heavy loads.

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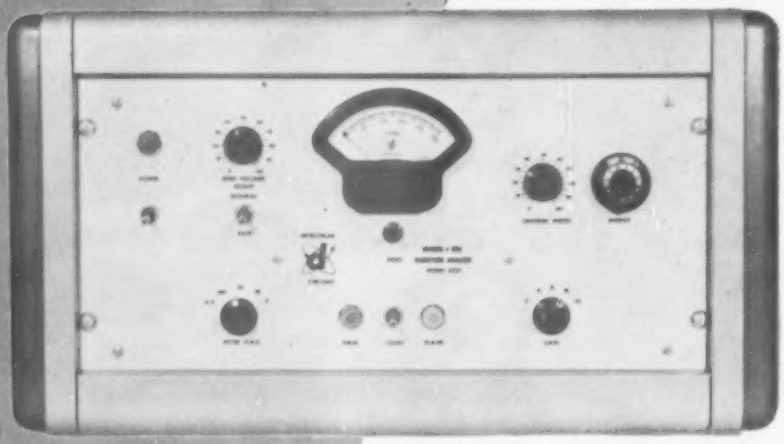

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THE DEMANDS OF INDUSTRY and science for *reliable* nucleonic research instruments is keeping the drafting boards at Detectolab busy with design activity. New instruments such as the above DZ21 Medical Spectrometer are urgently needed for the headlong race of science over problem barriers.

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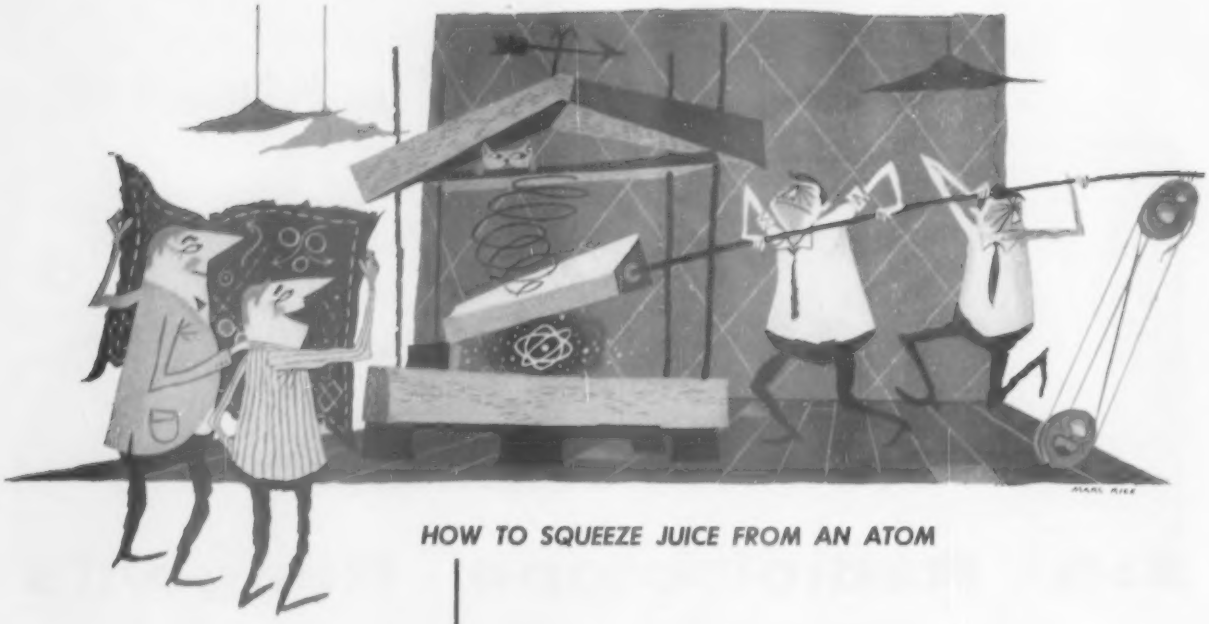

HOW TO SQUEEZE JUICE FROM AN ATOM

Detroit Edison Reactor Going Forward;

NDA - Nuclear Engineers

The AEC announced recently that the project "proposed by the Detroit Edison Company and Associates... had been determined by the Commission to give promise of significantly advancing power reactor technology and providing an acceptable basis for negotiation under the Power Demonstration Reactor Program."

"Detroit Edison and Associates propose construction and operation of a fast breeder reactor plant with 100,000 KW of capacity to be completed in late 1959. The plant is to be located within Detroit Edison Company's service area."

Nuclear Development Corporation of America — (NDA) — has had the nuclear engineering responsibility for the Detroit Edison project since its inception in 1950.

We will be happy to meet with engineers and scientists interested in participating with NDA in this expanding field.

NDA_

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ROUNDUP from GENEVA August 8 to 20

Editor's on-the-scene Commentary on Geneva Conference and Exposition

History was made last month at Geneva when the first world-wide discussion of peaceful atomic energy was held. United-Nations-sponsored International Conference on Peaceful Uses of Atomic Energy brought together almost 2,000 representatives of 72 nations to hear about 450 papers on almost every facet of the subject (papers not presented orally brought the total to over 1,000). In addition, there were sizable governmental and commercial exhibits. However, perhaps most important were the corridor talks among delegates—these provided basis for real contacts and information exchange.

Basic impact of Geneva on relationships between peoples and progress in atomic energy can be seen and evaluated easily. But detailed repercussions and long-term effects will be determined only after careful postconference study. This special Roundup report written in Geneva presents an over-all view of the conference.

Beyond question, participants agree that the meeting was an unparalleled success. In fact, there may never be another technical meeting with such a profound effect on technological progress and international relations at the same time.

ROUNDUP

FIVE MAJOR ACCOMPLISHMENTS of the Conference stand out clearly: • Lines of communication between East and West have been reopened in the sensitive atomic energy field—and these may force information exchange in other kinds of technology • It has been sharply demonstrated that technologists around the world think and act similarly though they work independently • Boundaries of where we stand today in atomic energy have been defined • Solid foundation has been laid for international commerce in atomic energy • And underlying fear of nations that spread of use of fissionable material may call for some kind of international control has begun to come to surface

Four categories of countries were represented at the Conference: Advanced nations (US, UK, USSR) with major nuclear programs, who gave out most information during meetings; Canada and western European nations now possessing or in process of building reactors; countries well along in initial planning of programs; and nations trying to find out what, if anything, they should be doing in atomic energy.

Despite wide subject range covered in program, far and away greatest interest was in reactors for power and for research. Each country in first category was anxious to give good showing in competition with other two nations. Those in second group were looking for specific nuts-and-bolts information that would help them in their work in progress. Those in third group were trying to confirm their ideas on reactor types to build. And last category was learning ABC's of atomic energy.

In general, little startlingly new information came out. But some details were filled in as each nation gave a rather comprehensive picture of what its experiences had been. This provided basis for more detailed corridor talks and frequent private-group get-togethers where real benefits of conference were realized. In fact, many delegates spent almost all their time outside meeting rooms. As far as US scientists and engineers are concerned, they came into the Conference not expecting to learn very much they didn't already know—and this is how it turned out. What they did get was a better view of other nations' programs, plus confirmation of some US data.

Each country had its own idea of possible significance of Conference to it both scientifically and politically and came to Geneva to get the utmost from it. Many delegations held detailed briefing sessions each morning to set up plans of attack for the day. US probably had most elaborate and carefully planned program. To handle its almost 350 people in Geneva, US set up replica of AEC Washington headquarters at classy duRhone Hotel. There key Washington staffers with their Washington secretaries operated in restricted area offices.

Obviously US went all out to put on a good show and did—by having its top people in Geneva presenting the most information on the greatest variety of subjects. Also, without question US official exhibits far surpassed any others in terms of information and workmanship.

Fusion. Most dramatic subject raised at Conference was controlled thermonuclear power. Because of coyness of Conference Big Three, it was not on agenda, but was brought up on opening day when Conference President Bhabha of India predicted thermonuclear reactors will be possible in two decades.

The Bhabha speech was then used by reporters at separate press conferences with five countries—US, UK, USSR, France, and India—to force admission that controlled-fusion programs exist in these countries. None would comment on actual work being done. Thus, there is still no official information available from any country on how important the work is, what the prospects are, or what competitive situation with fission is likely to be. However, at his press conference, Bhabha did offer some speculation on the nature of the problem. He said that temperatures of the order of a million degrees will be needed and that range of interest lies between several-thousand-degree temperatures produced by hollow charges and temperatures produced by low-energy accelerators. Accelerators are needed that produce several million electron volts at several amperes current. Reaction would probably be deuterons on lithium. Bhabha says reaction would start easier with Li^6, but would also work with Li^7.

Russia. Without doubt the biggest single interest among Western nations at the Conference was in seeing what the Russians were like and what they would say. Apart from a July atomic energy conference in Moscow, which seemed to have been arranged at last minute, Geneva had first public descriptions of Soviet nuclear projects. Two reactions to Soviets stand out: their scientists are basically no different than ours—technical people approach technical problems similarly the world over; and they have a program that must be reckoned with seriously in peaceful atomic race. Several members of Joint Congressional Committee on Atomic Energy even went so far as to place USSR second to the US (ahead of UK). But this does not appear to be borne out by material presented.

Flavor of Russian attitude at scientist level was perhaps best exemplified by remarks of Soviet chemist Vinogradov at press conference. He referred to speakers from other countries as, "my colleague radiochemists," and said that one session "allowed us for the first time on such a large scale to share information of chemistry of fission products." At one point he compared a Soviet exhibit with "remarkable progress" in US. He also said, "I've learned quite a bit thus far in my own specialty, radiochemistry. You miss information when you try to get it only through reading. Less is missed in personal contacts." He readily admitted coming across at Conference more sensitive method of polarographic analysis than he'd been using.

Reactors. Papers on reactors and private discussions provide good basis for comparison of US, UK, and USSR programs. It is interesting that there are more similarities between US and USSR efforts than between US and UK. This is probably because US and apparently USSR have adequate plutonium, whereas UK doesn't, and so is concentrating on dual-purpose plants.

Basic difference between US and UK was emphasized at Geneva. British need nuclear power badly (and need to reduce sterling spent importing fuel), and so are hitting one line of development hard—CO_2-cooled graphite-moderated reactors. They expect their requirements in year 2000 for equivalent of 250-million tons of coal will be met by burning 250 tons/yr of U or Th.

Calder Hall design was apparently originally intended to be power only, but was shifted to dual purpose recently. Taking reasonable fuel credit for plutonium, they say they can get about 7-mill power. One AEC official points out that same plant in US would produce 12–13-mill power because they use 8% for fixed charges whereas we use 15%.

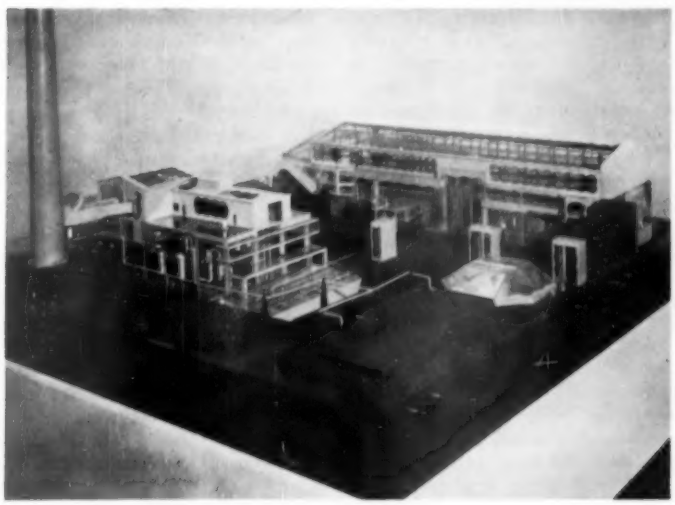
Although there is no doubt that US is considerably ahead of Russia in reactor field, US people expressed respect for their program. One thing is sure—we didn't learn anything appreciable from them. Two minor ideas did come out: use of eccentric cores for good neutron utilization in research, and avoidance of use of coffins for removal of fuel elements by essentially using reactor room as shield.

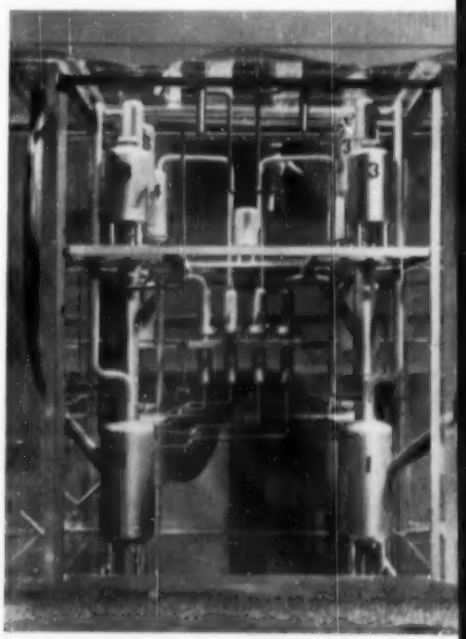
Russia is working primarily on graphite water-cooled reactors and will build larger ones of this type. Their 100-Mw plant is said to be scheduled for completion next year. They don't seem to have too much experience on other types. For example, in exchange with W. H. Zinn (of Argonne) after he gave paper on boiling reactors, they doubted such plants could be stable at high powers. Zinn said there are practical designs that go up to 1,000 Mw heat. Also, they showed design concept for boiling homogeneous reactor that had not been carried very far. For example, they talked about fission-product separator with removal factor of 10, but they really hadn't done more than check it on small lab scale.

US competitive position. Although US officials emphasized before Geneva that Conference was not being looked at as an atomic Olympic Games, there is now little question that it was. In fact, at one press conference, JCAE members turned the tables on the press and asked them questions about how US was stacking up against other nations. Regardless of criteria used, US leads in power-reactor technology and in prospects for developing foreign business. Although presentation of papers was basic purpose of Conference, official and commercial exhibits also were vital to success. Actually, for many foreign prospective buyers of equipment, this was the most important part.

US spent well over a million dollars on its exhibit and had star performer of the Conference in the swimming-pool reactor put up on UN grounds.

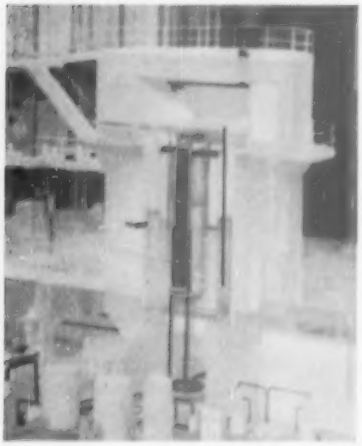
More on page 14



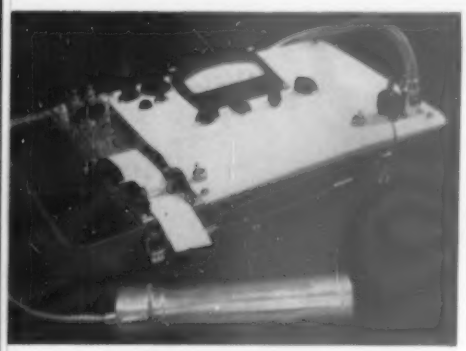


US's IDAHO Chemical Processing Plant model [at left] shows main process building with separate cells (rear), equipment for handling liquid and gaseous wastes (center) and storage tanks (front). A simplified cell model [above] shows: (1) dissolvers, (2) coarse feed adjustment, (3) filter tanks, (4) fine feed adjustment

ROUNDUP



HEAVY-WATER research reactor produces 0.5 Mw heat, uses natural U, is moderated and cooled by D_2O (see pp. 51, 53)



MEDICAL SCINTILLATION counter determines blood circulation by tracing radiosodium; uses crystalline CsI (Tl)

THERMAL RESEARCH reactor (TRR) will produce 2 Mw heat, will use 10% U^{235}, and will be moderated and cooled by H_2O (see pp. 51, 53)

LARGE CROWDS were attracted to Russian exhibit. Most displays related to measurement of radiation; three reactor models were shown

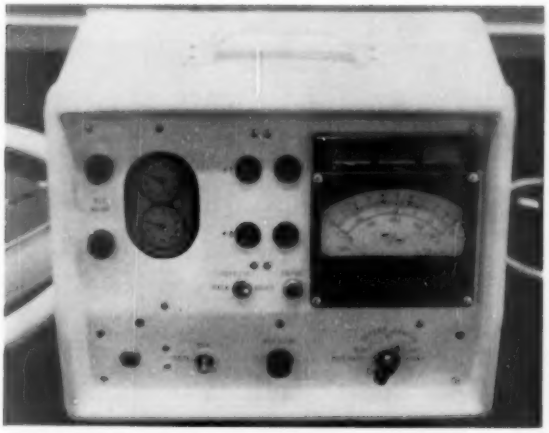
Russian Exhibit

Of official exhibits at the Palais des Nations, the Russian display was second to US's qualitatively and quantitatively. But on curiosity value, it ranked first. Reactions to it were mixed. Some felt their equipment displayed good workmanship. Others felt it was shoddy. In many cases, evaluation was difficult because it was impossible to get complete details on equipment design, operation, and performance. According to which Soviet guide you talked to, what piece of equipment you were asking about, and what time of day it was, you got more or less data.

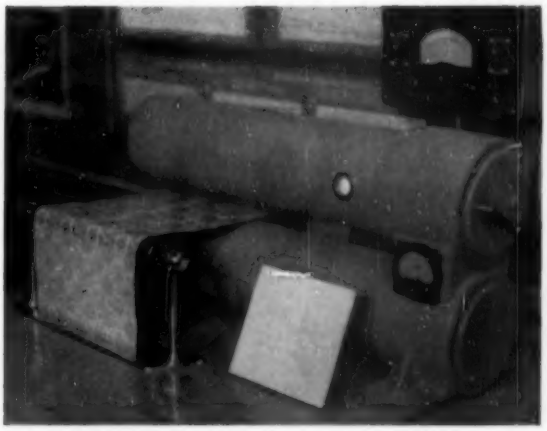
They showed one device that seemed to force use of radiation–bottle counter wherein bottles were counted by interruption of radiation beam. Photoelectric device would seem to have sufficed here.

Instruments were generally well put together, although over-all they seemed several years behind US on techniques. They showed a dual-head medical scintillation counter for following blood circulation using radiosodium. This used only photomultiplier they displayed here—one that RCA people say is copy of early RCA photomultiplier; it also has pinched neck (copied after EMI version) to reduce afterpulsing caused by ionization in residual gas. It has gain of about 10^6, sensitivity of 40 amps/lumen, applied voltage of 1,000–2,000 volts; they say it uses cesium iodide crystals bonded to tube by something like Canada balsam. They also say they're working on stilbene and terphenyl.

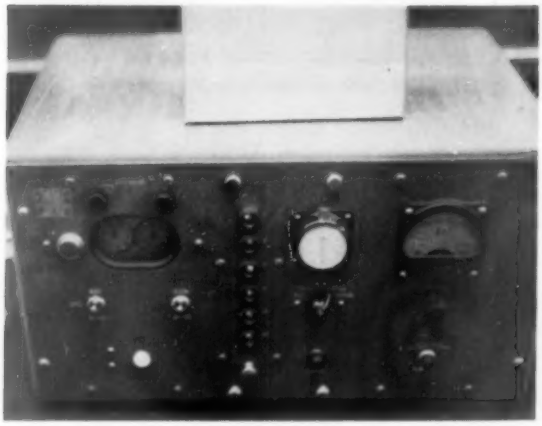
They showed neutron and gamma well-logging device for work down to about 6,000 ft. Detection head uses photomultiplier tube with



ROENTGENOMETER indicates total count or counting rate



BETA GAGE measures 100–700 gm/m² thickness with ±1% accuracy



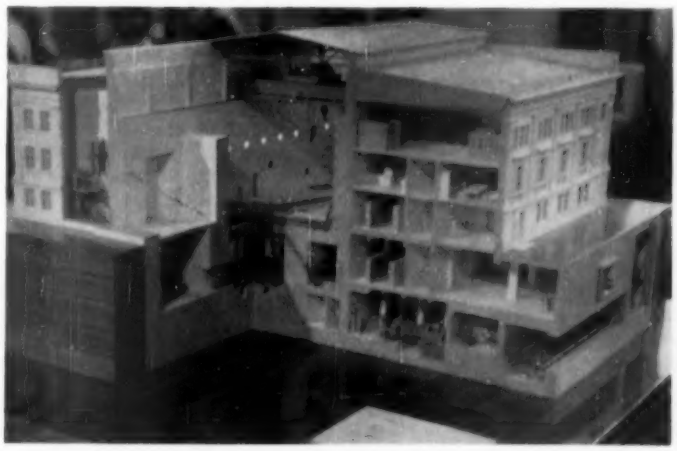
COUNTER UNIT has built-in 2.5-kv supply

TUBES for counter unit

Ranked High on Curiosity Value

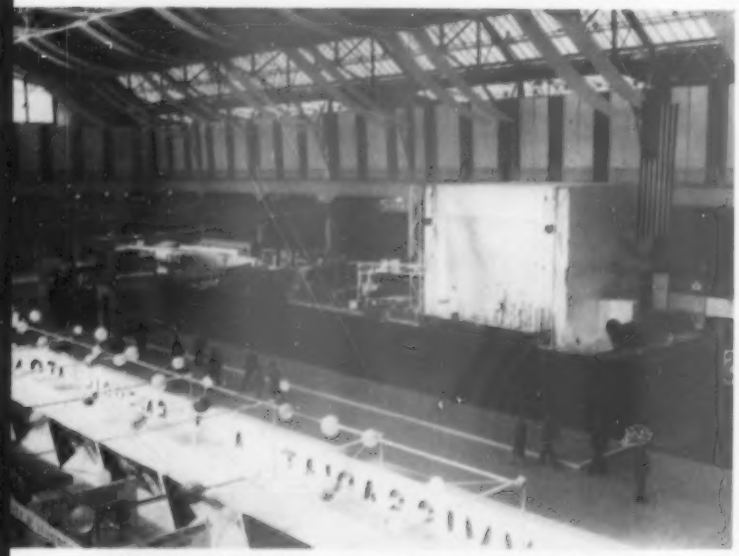
sodium iodide crystal. Electronic components were well mounted in probe, but they mostly used large vacuum tubes. In general, miniaturizing techniques displayed do not seem far advanced.

One interesting tube shown was a corona discharge alpha counter that is not triggered by betas and gammas, and thus discriminates against them. Tube operates on 400 volts, giving 50 volt pulse, and uses metallized mica window of 2 mg/cm². They also showed: a windowless flowmeter for soft betas in which they use Teflon and fluorothene liberally and use helium gas; a 4π counter; a gamma survey instrument reading 15 mr/hr full scale with 20% flat response from 70 kev to 2 Mev with an ion chamber with graphite-coated Lucite; and 200-mr pocket dosimeters that seemed similar to Argonne design.



APS-1 5-Mw reactor power station uses 5% U^{235}, graphite moderator, and pressurized H_2O coolant (see p. 40)

ROUNDUP



ORNL PILE model (US) and over-all view of commercial show

REACTOR FUEL elements in the US exhibit

It made profound impression on many thousands of visitors. At commercial show, US had largest popular display, but because of UK Government support, British companies outdid US industry on their exhibits.

Prior to Conference, general impression was that British industry had stolen march on US companies. Now, if any advantage exists, it is not so clearcut. One difference between the two countries that does stand out is working relationship between Government and industry in England and lack of same in US. This difference was emphasized at commercial exhibit where visitors to UK section were greeted by large sign reading "Complete Atomic Power Stations under Single Contract by Three Famous Firms (English Electric, Babcock & Wilcox, Taylor Woodrow) Working as a Group . . . With Active Encouragement and Close Technical Cooperation of Atomic Energy Authority." In US, the atomic energy law stipulates firms must clear with AEC before having any dealings with foreign countries. In practice, most companies have been ignoring this. But lack of AEC rules on this has been psychological deterrent to some.

Another advantage thought to exist for British was more liberal interpretation of declassification rules to facilitate discussions with foreign countries. In actuality, both countries have some basic declassification principles. But more important, with recently adopted liberal policies, companies in both countries now say that they can go as far as they need to in their sales pitches and that classification is no barrier right now.

To give US position a boost, AEC dramatically announced on Conference opening day a surprisingly low price for heavy water of $28/lb (p. 110). This is about one-third the Norwegian price. Norwegian sources doubt that this is a truly commercial figure and say it will prove embarrassing to Norway because of existing commitments there for next five years with Sweden, France, Italy, India, Yugoslavia, and Holland. AEC Chairman Strauss says this is the breakeven price. But another AEC official said D_2O would now be used as an item of international barter in same way as uranium. And it will be effective, because, unless Norway cuts its price, the many countries needing heavy water will turn to the US.

Another factor in the US-UK A-race is difference in reactors offered. British firms maintain they're in better competitive situation because technology of the plant they're selling is known and little development is required. Also, they feel the large sizes they're working on permit them to market what's needed. On other hand, US people say British have only one plant to offer—Calder Hall type—that it is not likely to be economical and that real need abroad is for plants of package type much smaller than Calder Hall.

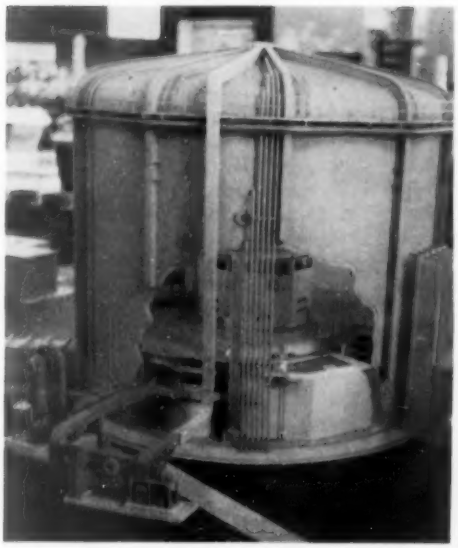
At Geneva, results of sales efforts bear out neither view conclusively. Only one known reactor sale was made—Westinghouse sale of 10-Mw enriched U pressurized-water package reactor, which was negotiated during Conference. It was announced by Italians, but not confirmed by Westinghouse. Although most reactor exhibitors proclaimed the show a success, no others could cite more than establishment of good contacts and the laying of groundwork with many customers. Most did not expect to do actual selling, although British were quoting prices and delivery dates (about $420/kw for delivery in about 1960, and that's higher than prices of some US firms).

There are no other competitors to US and UK on sale of reactors at present. However, USSR has said it is selling reactors to some of its satellites (p. 51) and it might make attractive offers to some neutral nations.

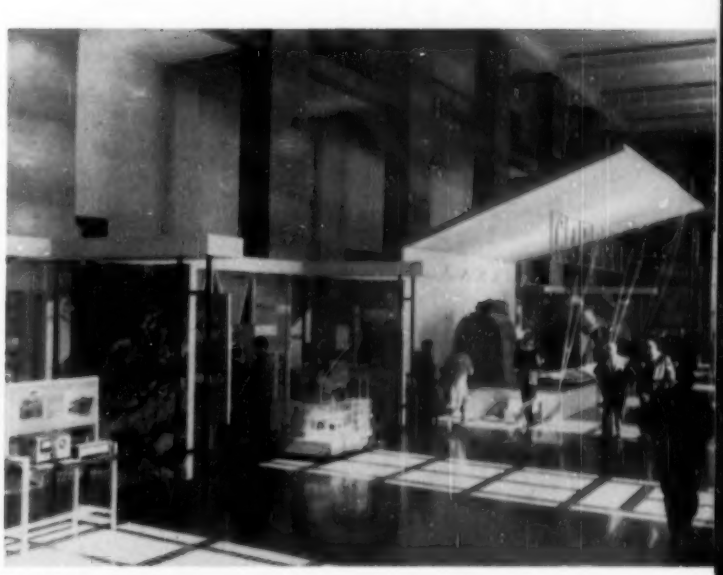
One problem facing both US and UK firms is that neither government has announced clearcut policies on making fuel available for power reactors sold abroad. Both have strongly hinted that this would be taken care of, but are perplexing their respective industries by not spelling out official views.

In US, bilateral agreements for exchange of classified information will be necessary before any power reactor deals can be consummated. Similar but less formal arrangements will be necessary in UK.

Secrecy. One major result that will flow from the Conference is a virtual complete breakdown of secrecy. This seems inevitable, particularly with realization that people everywhere recognize basic problems and ultimately come up with similar results. Laws of nature and tech-



CALDER HALL reactor model in part of British section



CANADIAN exhibit featured prospecting instruments, NRU reactor model, and two Co^{60} therapy units (one sold to City of Geneva, other to Marie Curie Foundation in Paris)

nological progress cannot be locked up in any one safe.

This conference saw the greatest flow of information ever produced at any nuclear meeting. Less-advanced nations were flooded with data. But it is apparent that we're now getting into age of industrial secrecy in atomic energy. Certain kinds of data were deliberately withheld at the Conference and undoubtedly will continue to be. England's Sir Christopher Hinton pointed this up when he was queried about details on their Calder Hall power reactor fuel elements. He said, "Quite frankly, I feel it is desirable to maintain a measure of industrial secrecy. One should realize industrial application of atomic energy is bound to be competitive. I think there is and should be competition. There won't be competition if we all disclose everything as soon as we learn it. I don't think we should have full exchange of information." AEC Commissioner Libby echoed this sentiment in commenting to NUCLEONICS on Russian papers, "We have no kick on Russian holdback on fuel technology. This is a trade secret. We do it too."

Thus little was presented by any country on fuel-element fabrication or performance. On the other hand, the US in its exhibit surprised everyone by displaying a variety of actual elements that could be examined closely by delegates (see also p. 58).

Another subject touched on lightly was technology of chemically recovering uranium and plutonium from spent fuel. Russia had nothing on subject. US described basic processes used and for first time showed general flow sheet of Idaho plant (see also p. 59 and 61).

FRENCH Commissariat a l'Energie Atomique exhibit

As usual, there is conflict in official quarters on where we go from here on secrecy. Senator Anderson, JCAE chairman, says there will be complete declassification as result of conference because "it isn't so important anymore to hang onto things we've been holding onto." But AEC Commissioners Strauss and Libby say we don't have to go any further. However, there already are plans to explore this in detail in Washington.

International agency. Pool first proposed by President Eisenhower in 1953 will undoubtedly be set up (proposed statute is scheduled to be published momentarily), but there is now little enthusiasm for it. In the US, there is still the same Congressional opposition that caused the crippling clause to be written into the atomic law. In answer to NUCLEONICS query on JCAE attitude to pushing signing of bilaterals versus developing agency, Senator John Bricker said, "We're getting along well with the bilaterals now."

Other countries feel they can now get what they want by playing ball with US and UK via bilateral agreements. One foreign delegate said he felt most countries realized the US was using bilaterals in part to keep these countries from dealing with each other. His country didn't mind

ROUNDUP

ITALIAN section included booth of National Committee for Nuclear Research

SWISS section—Brown Boveri and Co showed betatron that produces betas and gammas up to 31 Mev. Accelerating tube is at lower right

SCANDINAVIAN joint exhibit included Kjeller and Swedish reactor models

as long as they got what they wanted, which they seemed to be in process of doing. Thus, agency isn't being looked to to push peaceful atomic energy.

However, another angle is that East and West are afraid of world-wide spread of nuclear power. There is feeling that, as fissionable material becomes more widely available with growth of atomic power industry, there will be a need for international policing. One Britisher felt a function of agency would be to oversee gaseous diffusion and chemical separation plants.

In this connection, an international activity that is being pursued in Europe is the formation of a small group to set up such plants on the Continent so as to make its members independent of US and possibly UK. A fear expressed by one European observer is that the US might then do same thing with enriched U as it has with D_2O—sell U at a very low price based on plants amortized for other purposes.

Physics. Perhaps some of most dramatic developments took place in the physics sessions. Data on cross sections of fissionable materials that had been compiled independently in secrecy in US, UK, and USSR over last 10 years were laid out in open. Remarkable thing was agreement on results, particularly on low-energy cross sections of plutonium. Spearheaded by D. J. Hughes of Brookhaven, representatives from US, UK, USSR, and France agreed on "world average values" for thermal (2,200 m/sec) absorption and fission cross sections. Quoted errors are based on spread in reported data (first number given after isotope is absorption cross section, second number is fission cross section, both in barns) U^{233} (593 ± 8 and 524 ± 8); U^{235} (698 ± 10 and 590 ± 15); Pu^{239} ($1{,}032 \pm 15$ and 729 ± 15).

In reactor physics, USSR appears to be good on theory but not on refinements of it. For example, they don't seem to use computing machines, which perhaps also holds back UK. In response to US question in reactor session about how flux in fuel rod was determined, USSR said they calculated it via diffusion theory, which US people say cannot be accurately used for this.

In one paper, USSR showed slight difference in U^{235} fission spectrum. US people say this causes error in calculation of neutron age. And difference of age in D_2O affects reactor physics calculations. Russians would think their systems have more neutron leakage and this would cause them to be reluctant to build smaller reactor systems.

Radiation chemistry. USSR work appears to be of good quality. And they may have more people working in this field than US does. They have looked at a number of problems that we haven't, but are still in research stage, not having done much on applications. Some examples of work they've done that we've not: effect of radiation on ethanol and acetic acid; effect of radiation on reaction between ammonia and benzene to form aniline; effect of radiation on production of ozone in liquid oxygen; creation of hydrogen potential under influence of gamma rays on platinum electrode; and improved sensitivity of reactions by use of oxide suspensions, for example, oxidation of benzene.

USSR mainly has been using cobalt sources in pellets or rods of strength about 500–1,000 curies. They use point-source technique with sample on either side of source, and, thus, don't get homogeneous radiation field. For assay, they use ion chambers and ferrous-ferric systems.

One French paper on application was quite interesting. It described grafting one polymer to another to form copolymer that has better properties than either (for example, they claim they can graft acrylonitrile to vinylacetate to get copolymer that will not soften in boiling water: both ingredients are much cheaper than polyethylene). Two techniques for doing this were described: (1) put polymer in a monomer in which it would swell, giving point contact between two, and then irradiate swollen mixture to get grafted copolymer; (2) irradiate a polymer in oxygen field or air (author Magat theorizes he gets peroxide linkage that will break at elevated temperature); then put treated polymer in contact with monomer and you get grafted copolymer; he has done this with polyethylene that he irradiates and then puts in contact with acrylonitrile. He says this product can be written on with pen, which is great advantage for packaging material.

Chemical separations. This was first time various solvent extraction pocesses used in US have been described in public. However, French claimed to give most details on their plant going up at Marcoule (p. 61). Latter will process several hundred thousand curies per day; it is scheduled to start operation in October, 1956.

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Nuclear Radiation—Its Detection and Measurement

by Dr. Ernest H. Wakefield, Director, Radiation Counter Laboratories, Inc.

There are six different types of commonly used nuclear radiations: alpha particles, beta particles, positrons, gamma rays, protons, and neutrons. Each of these radiations is detected directly or indirectly by the ionization which it causes as it passes through a medium, such as a gas, solid, or liquid.

An alpha particle is a fast moving helium nucleus. This radiation is given off by several elements, among them plutonium, one of the components of the atomic bomb. Alpha radiation has typically a path length in air of from 2 to 5 cm; hence it can be stopped by even a sheet of paper. Alpha particles are emitted in a monochromatic or line spectrum and are characterized by a unique range.

In contrast, beta radiation, which is the name given to fast moving electrons emitted from radioactive isotopes, is emitted in a continuous spectrum of energies up to a maximum value. Beta particles have a typical path length in air from a few millimeters to perhaps a few tens of meters. Cosmic ray electrons, however, may have path lengths in air much greater. Beta particles generally can be shielded against by a thin layer of metal compared to correspondingly energetic gamma rays which are discussed below.

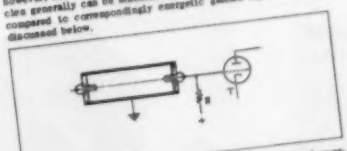

Fig. 1 - Ion Chamber. Physically, an ion chamber, a proportional counter, and a Geiger-Mueller counter may each be considered as a conducting cylinder which has coaxial with it a conducting wire known as an anode. If the instrument is used as a proportional or as a Geiger-Mueller counter, a condenser is placed between R and tube T.

Positrons, which are positive electrons, may have an extremely short life combining readily with free negative electrons, with which they are identical in regard to path length for a given energy.

Gamma radiation, which is given off by a host of isotopes and which is considered less particulate than the two previously named radiations, is absorbed exponentially, and hence there is no absorber thickness which reduces the intensity truly to zero. Therefore, shielding against gamma radiation is difficult. Protons, which have a path length of several meters in air, can be readily stopped, resembling alpha particles very closely in this respect.

Neutrons are unlike the previously named radiations in that they are non-ionizing in nature, but when they enter a nucleus often convert it to a radioactive isotope whose emission is detectable (for instance when they enter the nucleus of the boron 10 isotope, the nucleus disrupts, emitting an alpha particle and a lithium particle; the nucleus disrupts, emitting an alpha particle and a lithium particle; the alpha and lithium particles can be detected). So neutrons are detected by secondary means. They are easily absorbed by thin layers of certain elements such as boron and cadmium, each of which has a large capture cross-section for neutrons, when the neutrons are of low energies. Neutrons are extremely penetrating when their energy is large. Effective stopping of them is therefore achieved by first reducing their energy with a "moderator" such as paraffin or water. Fast neutrons may also be indirectly detected by observing the ionization resulting from the recoil-proton emitted by a neutron irradiated hydrogenous material, such as a plastic, or an anthracene crystal.

Sensing Elements

Eight different instruments are used for the measurements of these nuclear radiations. It will be strategic for us to discuss first the sensing elements of these instruments and then their recording elements, keeping in mind the principles underlying each type.

A natural listing of instruments for the detection of radioactivity would include photographic film*, conduction detectors*, chemical integrating indicators*, ion chambers, proportional counters, Geiger-Mueller counters, ion chambers, proportional counters, Geiger-Mueller counters, and scintillation counters. Each of these instruments is, in truth, a sensing element for ionizing radiation, but associated circuits may be required to record the intensity of the radiation that the sensing element detects.

ION CHAMBERS

An ion chamber is a volume of gas in which an electric field is created by a system of charged electrodes. Ions produced in this gas by an ionizing event are collected at these electrodes. The ion current is a measure of the character and intensity of the radiations. The gas amplification within the counter tube is unity; hence the signal from the ion chamber requires considerable amplification before it can be recorded. The anode is generally 1/10 to 1/100 the diameter of the cathode and is coaxial with it. Figure 1 illustrates the principle of an ion chamber. A typical voltage across these electrodes would be 100v. Ion chambers are widely used as instruments in health work, because such as they establish to the highest degree a linear relation-

Fig. 2 - The Nucleometer - Mark 9 Model 6A - as a proportional counter

*Further details concerning these sensing elements may be had by writing the author.

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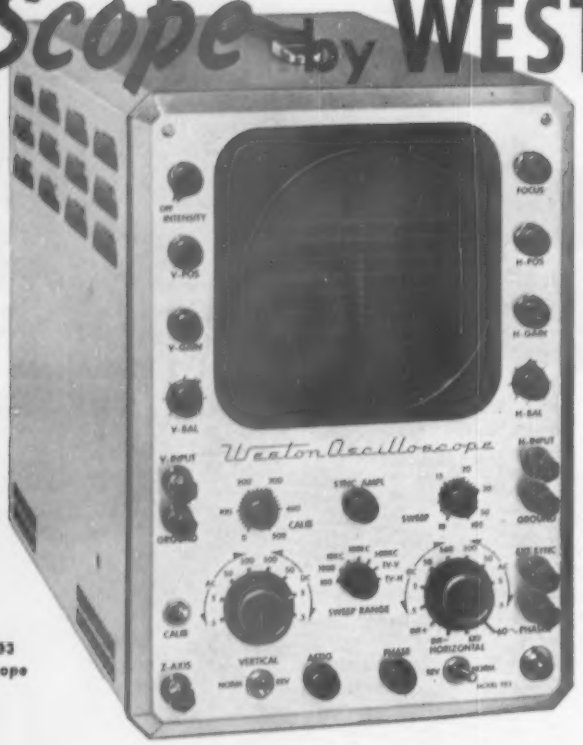

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Model 983
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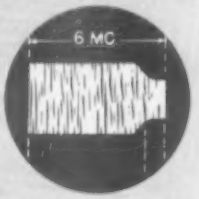
RESPONSE CHARACTERISTIC



Note flatness throughout specified range; to 3.6 mc down 1.5 db, at 4.5 mc down 3 db, at 6 mc down 6 db.

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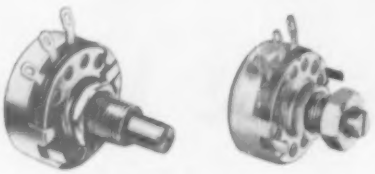

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Type T potentiometer

Type SO stand-off capacitors

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Type FT feed-thru and Type SO stand-off discoidal capacitors exhibit no parallel resonance effects normally encountered with tubular capacitors in VHF and UHF frequency ranges.

Type FT feed-thru capacitors are furnished with soldering tabs or with screw thread mountings.

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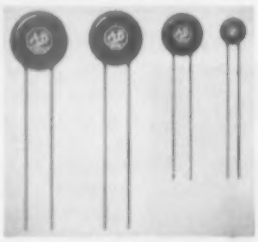

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In addition, Norton is presently developing numerous experimental items. The Norton research and development organization includes an engineering staff experienced in atomic development — available for cooperation where electric furnace products may be of interest.

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Above: Norton arc-type furnaces. *Below:* Norton resistance-type furnace.

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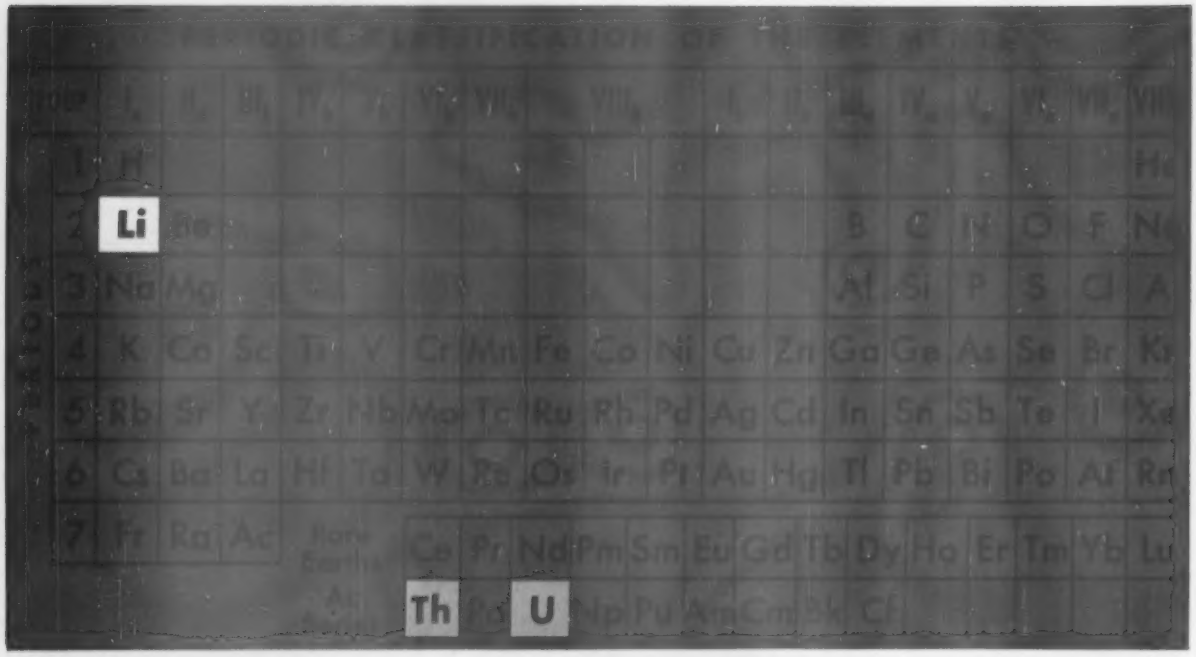
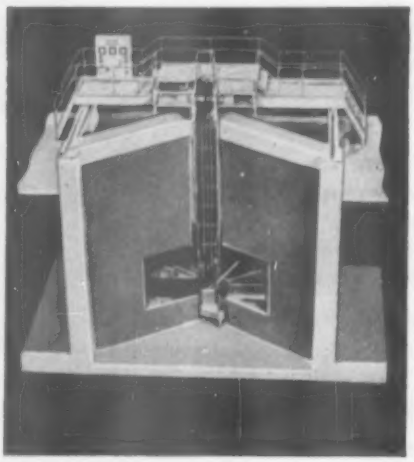


LITHIUM in the ATOMIC FUTURE

Only in recent years has mention of Lithium in nuclear energy been made. For the past 12 years, however, the technical staff of Lithium Corporation has been accumulating know-how on lithium. The contributions made to the comparatively small store of knowledge on this interesting element are material. Now, with our country on the threshold of peacetime nuclear power, Lithium Corporation stands ready to assist the creators of tomorrow's power. Lithium metal, two metal derivatives, and 15 salts are available for study. You are invited to share the benefits of our use-research into lithium. Write for technical data sheets on this miracle element. It could mean a great deal to you.



. . . trends ahead in industrial applications for Lithium

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Advances in Applied Radiation

DEVELOPMENTS in the FIELD OF APPLIED RADIATION ENERGY, its APPLICATIONS and the APPARATUS USED TO PRODUCE IT

High Voltage Opens West Coast Office

To handle sales and to provide service to customers in the Western part of the country, High Voltage has established a West Coast office at 1644 Grove Street, Berkeley, California. Manager of this new activity, which will also engage in the development of advanced accelerator systems and components, is George C. McFarland, for the past nine years a physicist at the Radiation Laboratory, University of California. Associated with him will be a group of physicists and other scientists having extensive experience in particle-accelerator design and construction.

High Voltage, the only commercial builder of Van de Graaff accelerators, has recently announced a newly developed line of microwave linear accelerators. Both types of machines produce intense beams of radiation which have wide application in industrial radiography, chemical processing, "cold" sterilization, medical therapy, and nuclear research. Many industries of the West — aircraft, chemicals, oil, rubber, plastics, and shipbuilding — have use for such particle accelerators and can now be served directly by High Voltage's new office.

Middle East's First Accelerator

The Weizmann Institute of Science in Rehovoth, Israel, has ordered from High Voltage a 3-million-volt Van de Graaff positive-ion accelerator for use in a program of atomic energy development now being carried on by that country. The machine will be the first "atom smasher" in the Middle East.

Dr. Amos de-Shalit, head of the Institute's physics department, plans to use the accelerator in the study of nuclear reactions produced by the bombardment of nuclei with high-energy protons. So used, Dr. de-Shalit says, the machine will contribute substantially to his nation's atomic research program.

Shell Buys Powerful Accelerator

Shell Development Company has just purchased from High Voltage a 3-million-volt Van de Graaff electron accelerator for radiation processing. This machine — the most powerful radiation source available to industry — is to be installed at Shell Development's Emeryville, California, laboratories. Shell scientists will use the accelerator in studying the effects of radiation on fuels, lubricants, plastics, and other oil-derived materials. The program will be aimed at learning more about changes that take place in matter exposed to high-energy radiation. Such information could lead to the creation of entirely new products, Shell's experts believe.

Prime advantage of this new accelerator is the extreme intensity of its radiation field. This intensity is several hundred times that of the most powerful radioactive cobalt source (4000 curies) now used by industry, or of any proposed for industrial radiation in the future. *In fact, this machine can process more product per unit of time than could the combined outputs of all the radioactive sources in the United States.* Shell's scientists state that the unit can treat more plastic in one hour than other radiation processes can in twenty days.

Other features are its easy adaptability to continuous, or conveyor-system, production techniques and the fact that it presents no radiation hazard when it is turned off. This new accelerator, which will cost about $117,000, will be delivered early in 1956.

Shell Development also has a 2-MeV positive-ion Van de Graaff installed in its Houston laboratory. Researchers there use the machine in "aging" rocks atomically — by bombarding them with protons, producing in one week, they say, the amount of radioactivity to which a rock might have been exposed during ten million years, in nature. Such studies as these can help scientists determine the age of rock formations and thus provide new clues in the search for oil.

Shell's 3-MeV Accelerator

Six-Mev Van De Graaff to Columbia

The huge six-million-volt Van de Graaff particle accelerator purchased by the U. S. Atomic Energy Commission for installation at Columbia University in New York recently passed all of the rigid factory tests required by High Voltage, and shipment to Columbia was completed in August.

The accelerator will be used primarily in the measurement of basic nuclear properties and neutron cross-sections.

Spain to Study Light Nuclei

The Junta de Energia Nuclear of Spain has received its 2-million-volt Van de Graaff positive-ion accelerator from High Voltage. It is to be used in a study of light nuclei, under the direction of Dr. Carlos Sanchez del Rio. The machine is located at the Centro Nacional de Engergia de la Moncloa.

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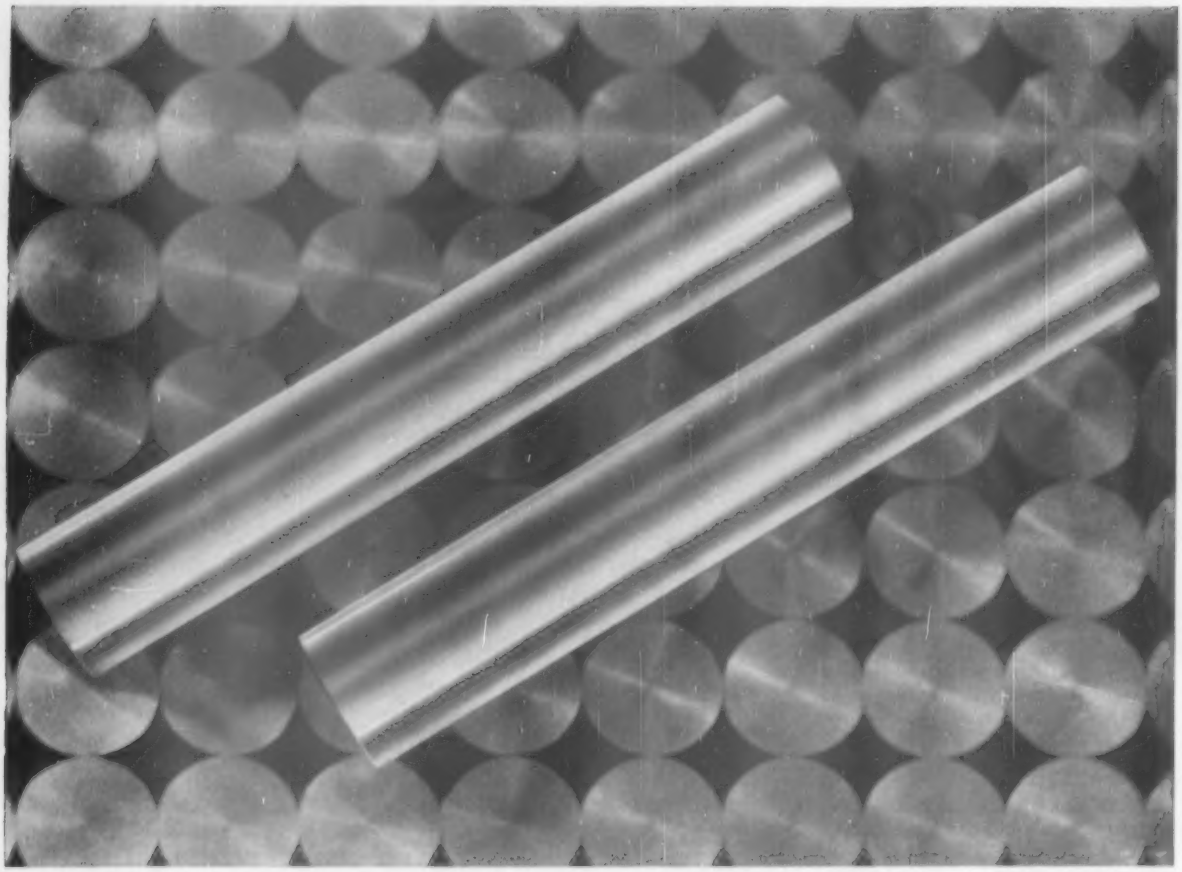
See our Exhibit at the U.S. Trade Fair of the Atomic Industry, Washington, D.C. Booth 625.



Atomic reactor fuel elements—a few of many thousands produced by Sylvania's nuclear fuel manufacturing facilities, Hicksville, N. Y. Fuel shown ¾ actual size.

10,000 tons of coal . . . 1955 style—

When the atomic energy power plants now in planning are completed, *reactor fuel* may become as important economically as coal, gas or oil is today.

Recognizing this, wise planners in atomic energy are investigating reactor fuel sources *now* . . . discovering that Sylvania has long been a leading producer of nuclear fuel elements and reactor components.

Sylvania's Atomic Energy Division is thoroughly equipped to supply production quantities of standard atomic fuel elements or special designs to meet your particular requirements. In addition, Sylvania will work with you in developing effective, economical fuel reprocessing methods so that your over-all costs will be as low as possible.

If your power or research reactor is still in the planning stage, you are invited to take full advantage of Sylvania's years of pioneering experience in the design and manufacture of atomic fuels and specialized components such as pumps, valves, test loops and reactor control elements. You will benefit from the combined skills and experience of more than a hundred specializing scientists and engineers.

Whether you are operating, planning, or just thinking about power reactors—Sylvania's scientific and engineering staff will gladly discuss your problems with you. For further, detailed information including a descriptive brochure of Sylvania's atomic energy activities, call or write:

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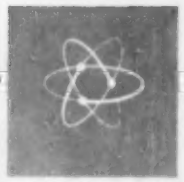

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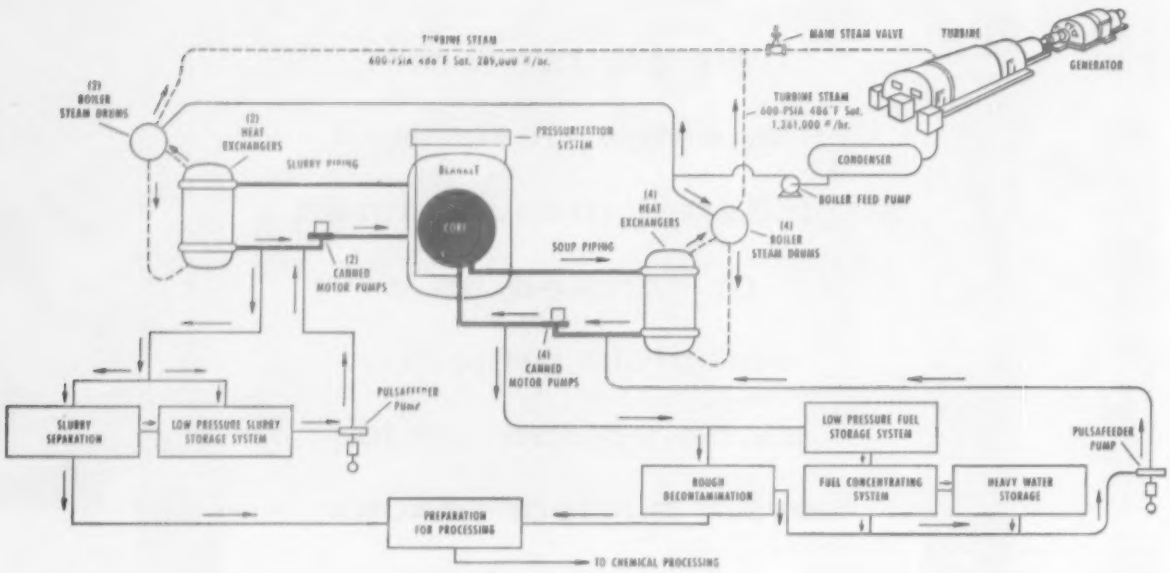
with new Aqueous Power Breeder



100,000 kw plant, at estimated cost of $21,000,000, will produce electric power at 6 to 9 mills per kw-hr!

The Aqueous Homogeneous Power Breeder type of nuclear reactor, heretofore regarded as a "long-term" possibility, is *now being offered* by Foster Wheeler as a practical solution to the problem of high construction and kw-hr costs of atomic power plants.

On the basis of design data worked out by Foster Wheeler, a 100,000 kw plant can now be built at an estimated cost to the owner of $21,000,000. With the Aqueous Homogeneous Power Breeder, electric production costs could be as low as 6 to 9 mills per kw-hr — depending on system load characteristics, plant location and method of financing. The estimated cost includes fuel, fertile material, reactor and turbine plant, buildings, shield, site preparation and land.

The Aqueous Homogeneous Power Breeder System has many inherent advantages. It utilizes a liquid fuel, is self-regulating, and increases safety by providing for the continual removal of insoluble fission products.

Simplicity of operation and accessibility of the equipment are among the unique features of the plant. Although designed for minimum maintenance, every piece of equipment is readily accessible for maintenance or repairs without excessive radiation hazard.

Steam produced by the reactor system is used to run a conventional turbo-generator.

For further information on nuclear reactors, write to: *Foster Wheeler Corporation, 165 Broadway, New York 6, N. Y.*

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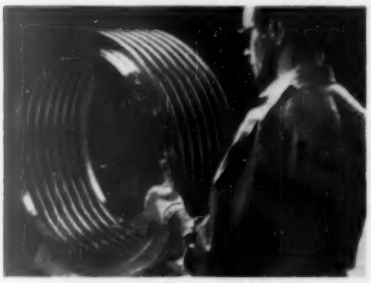
APPLYING ATOMIC POWER?

Call on Solar skills with special metals for design and production of nuclear industry components

BUILDING FOR THE ATOMIC AGE calls for many special skills. For 28 years, Solar has been solving the problems that today confront nuclear engineers—extreme pressures, intense temperatures, corrosive conditions. Solar overcomes these hazards with highly precise fabrications of stainless steels, super alloys, titanium and other metals. This knowledge—expressed both in top quality contract production, and in the varied Solar proprietary lines—is now available to business firms and government agencies in the nuclear field.

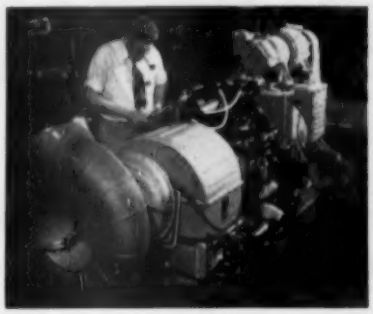
SOLAR BELLOWS for an AEC installation are shown below. They are one of several components being built currently for nuclear processes. In production they demanded strict surface perfection. The stainless steel nacelles pictured at bottom call for the expert forming and exact welding of half and full hard stainless steels.



Surface perfection required for AEC use

CURRENT PRODUCTION at Solar also includes pneumatic ducting, aircraft engine components and varied hard-to-build items for the petroleum, chemical and power industries where many of the problems of nuclear processes have been encountered.

Solar builds first stainless nacelles



Jupiter gas turbine is light, compact

GAS TURBINE ENGINES designed and built by Solar include the 50 hp Mars® and the 500 hp Jupiter. Their simplicity, ease of maintenance, high power output in relation to size and weight, and versatility make gas turbines the ideal prime mover for many specialized applications.

PROTECTIVE COATINGS developed by Solar have proven useful in extending the service life of alloy metals in cases where intense temperatures and highly corrosive conditions previously presented design limitations.

SOLAR OFFERS MORE than metal working skills and special products for atomic installations. A comprehensive research organization, and design, development and production engineering groups supplement manufacturing facilities. Solar research and engineering "firsts" include the first practical jet engine afterburner, the first hand started gas turbine engine, the first proprietary line of ceramic coatings for high temperature service, and many others dating back to the first stainless steel aircraft manifold.

PRODUCTION FACILITIES cover every type of metalworking—forming, machining, welding, brazing, casting, coating. Equipment ranges from precision experimental and tool shop machines to specially designed high production units. Solar has designed and built many of its own welding and fabricating machines for specialized needs. Some have proved so useful that they have been ordered by firms in the United States and Europe.

Research facilities include 19 test cells

Engineering task force studies control devices

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LET SOLAR PROVE how its skills are "tailor made" for the production of nuclear industry components. Describe your requirements, and let a Solar task force demonstrate that Solar is your logical supplier.

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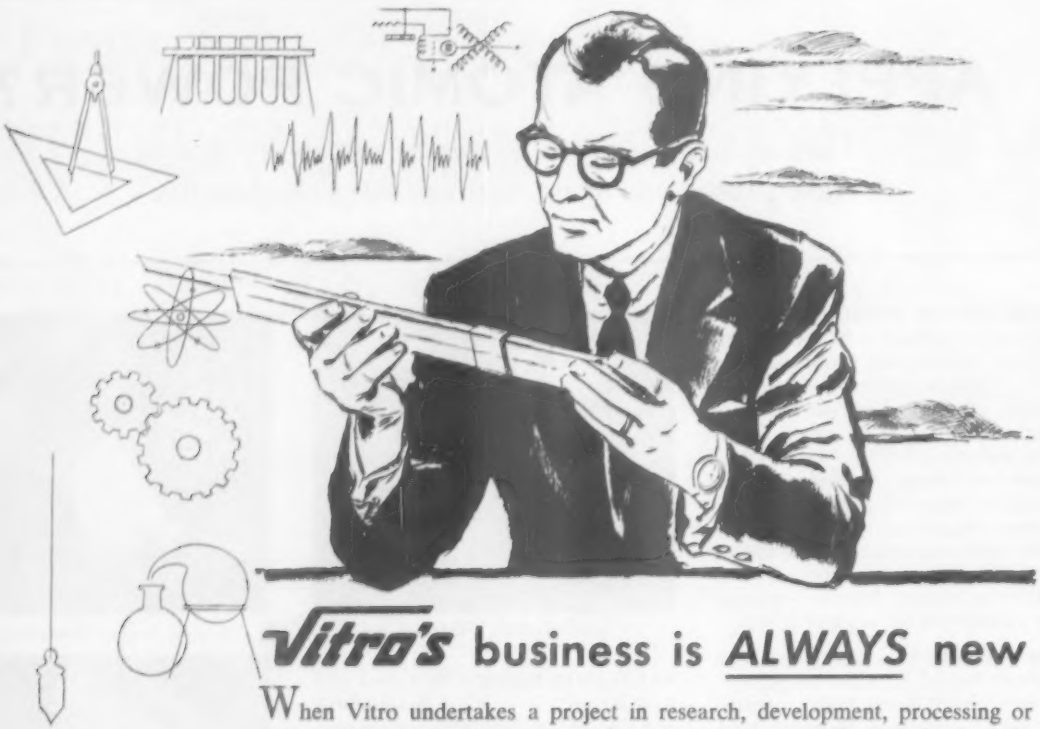
INFORMATION

For more information regarding any Solar product or service, write Solar Aircraft Company, Dept. B-55 San Diego 12, California



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- uranium mining, milling and refining processes
- electronic equipment reliability predictions
- rocket, guided missile and fire control systems

Vitro's interests continue in the solutions to industrial problems. With the tools of chemistry, electronics, nucleonics, metallurgy, processing and other technology Vitro's activities are increasingly valuable to the growth of the Atomic Age. On request VITRO will supply detailed literature.

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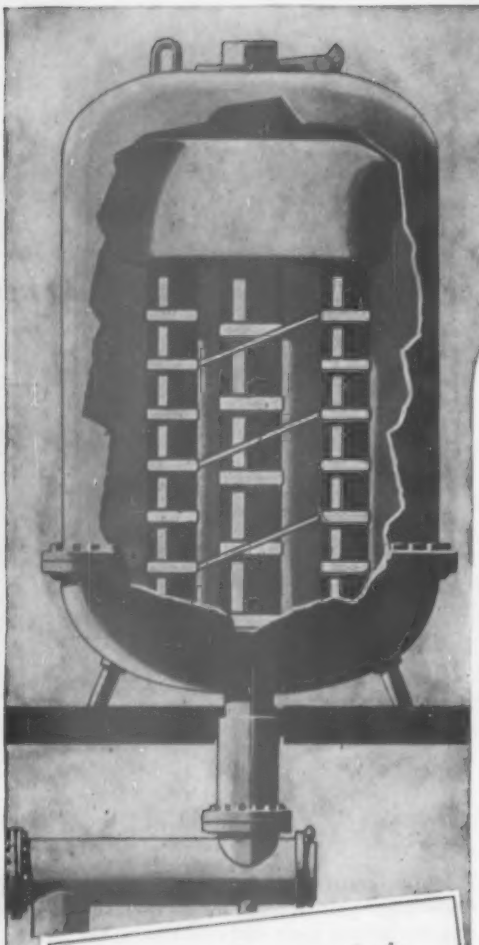

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LOW OPERATING VOLTAGE makes the G-E counter convenient for field as well as laboratory use. The proportional counter does not require the high supply voltage usually needed for this type of instrument. Because the pressure in the tube is low, the instrument will operate on 500-700 volts.

WIDELY APPLIED, the G-E neutron proportional counter is well suited for health-physics applications, oil-well logging, and the measurement of neutron fluxes in reactor installations.

FOR MORE INFORMATION, contact your nearest G-E Apparatus Sales Office or write for bulletin GEC-850, to General Electric, Sec. 585-21, Schenectady, N. Y.

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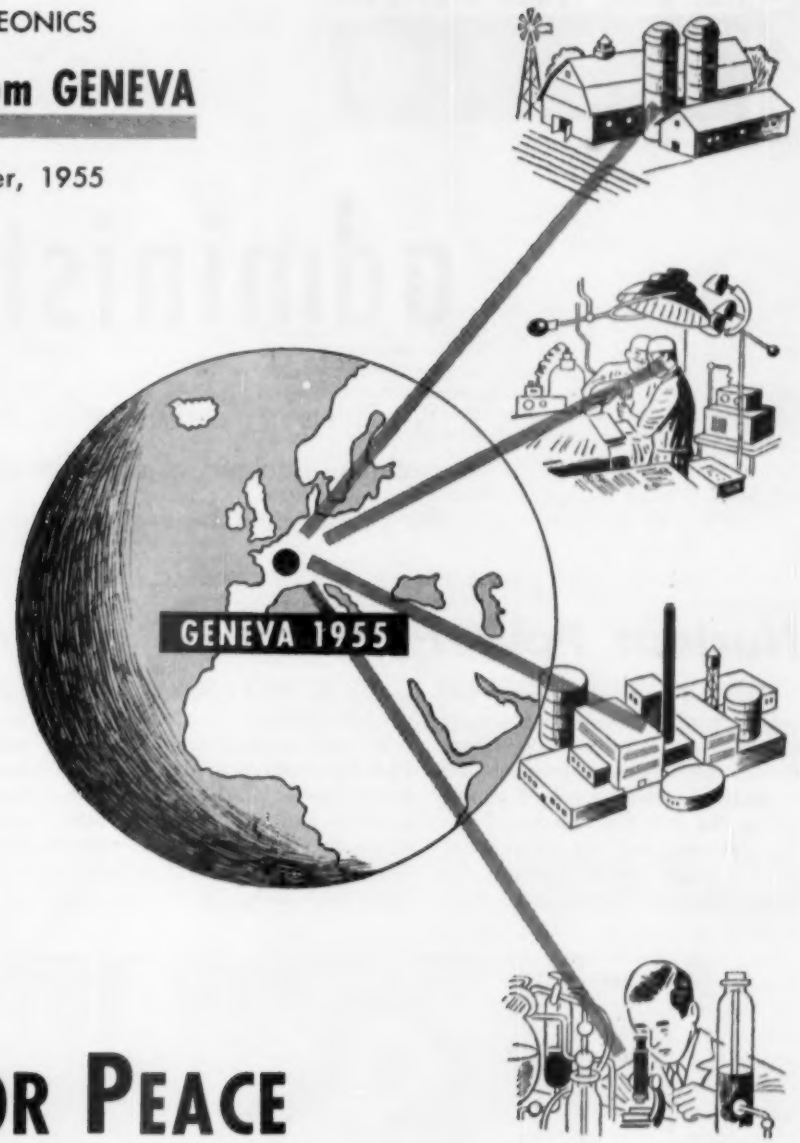
REPORT from GENEVA

September, 1955



ATOMS FOR PEACE

AUGUST 8 TO 20, scientists from 72 nations gathered at Geneva for the UN-sponsored International Conference on The Peaceful Uses of Atomic Energy. Their cooperative spirit in sharing information gives new hope for nuclear progress after a decade of duplicated effort and fragmentary information. In 60 sessions, they presented over 1,000 papers on all aspects of nuclear technology. Emphasizing new and significant data from the conference, the Editors of NUCLEONICS have written this special report to help fulfill the promise of the peaceful atom.

REPORT from GENEVA

administration

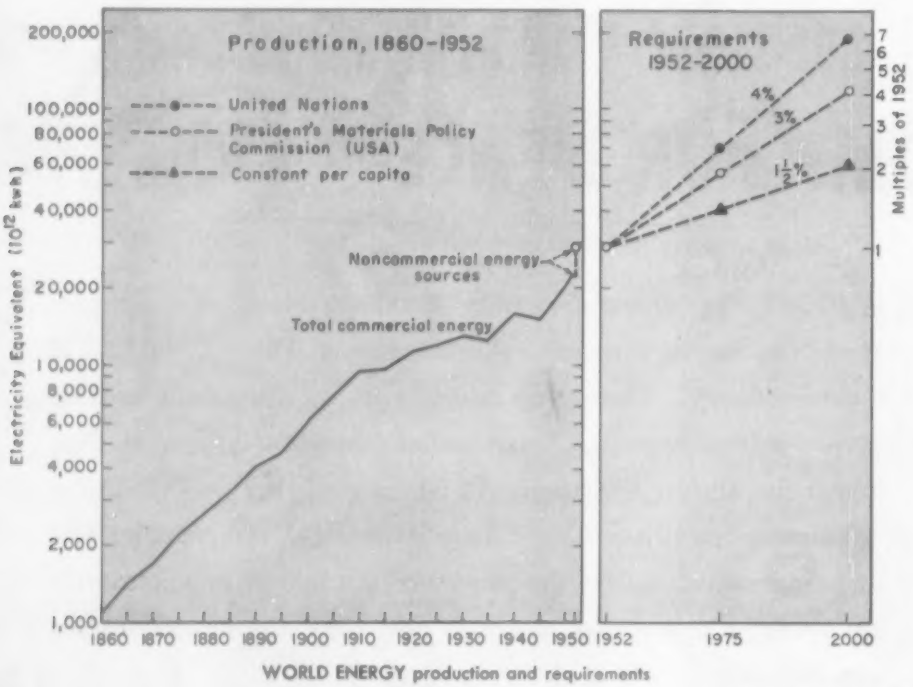
World energy consumption—its growth—energy-poor regions—nuclear vs established fuel costs—regulation for public health—reactor exposure cases

Nuclear Power—Need

There is a pressing urgency to find out how much nuclear power the world is going to need in the next half century. Enough conventional fuel sources exist to satisfy a customary rate of growth during the next few decades. Costs may increase, but not very much. Over a longer period, however, the picture changes. Nuclear power could become the be-all end-all of our economic well-being.

To make an accurate prediction of what the world will need in the way of nuclear power in 1975 and 2000 is impossible. But we must, just the same, arrive at some estimate of present demands and resources in relation to future needs, dwindling power reserves, and technical capabilities. The need will depend largely on: (a) how much energy will be required, (b) how economically nuclear sources can be tapped, and (c) the availability and adaptability of nuclear energy (*757, 802, 902*).*

* References are to UN papers listed on p. 94.



WORLD ENERGY production and requirements

Energy Requirements of Individual Countries

Country	Current energy consumption ($\times 10^9$ kwh) Year	Total	Electrical	Projected consumption ($\times 10^9$ kwh) Year	Total	Electrical	Comments
France (*346*)	1954		45				
India (*868*)	1953	1,308	6.7				75% of energy consumed obtained from cattle dung. Installed electric generating capacity: 1954—2,305,190 kw, 1986—50,000,000 kw. Coal reserves appear sufficient to permit max. industrialization during next few decades
Israel (*772*)	1954	10.6	1.1	1964	36	6.3	⅓ of generated electricity now goes for irrigation. Practically only energy source now is imported petroleum. Studying wind, solar, hydroelectric, and nuclear energy
New Zealand (*1082*)	1955		4.4	1975		16.5	
Pakistan (*1113*)							1955: demand = 5.5×10^8 kw; capacity = 1.3×10^8 kw 1975: demand = 19×10^8 kw; capacity = 8.4×10^8 kw
Republic of Korea (*1115*)	1954		0.9	1975		9	
Sweden (*781*)	1954	150	23.7	1975		75	Water power is 95% of '54 and 84% of '75 electrical consumption; will be fully exploited by '75. 70% of energy consumption depends on imports. Large deposits of low-grade U ore
Taiwan (*864*)	1954		2	1975		9.6	
United Kingdom (*388*)	1954	463	70	1975	704	220	Coal main source. By 1975, ¼ of primary fuel devoted to electricity generation
Yugoslavia (*962*)	1952	102		2000	590		By 2000, 70% of requirements will be filled by present energy sources

Nuclear Growth Factors

▶Demand for energy will increase.

▶Increase in requirements is rapidly diminishing known and estimated reserves of energy-producing fuels, despite increased production efficiency.

▶Urgently needed is a new source of energy.

▶Nuclear energy, if it is to supplant or supplement present energy sources, must be reduced in cost to a level comparable to present production costs.

▶There is a marked preference in the world for conversion of primary fuels into electricity before consumer use.

▶Rapid growth of nuclear power production will be spurred on by the increasing demand for electricity.

▶Supply of energy and energy consumption are themselves among the principal factors influencing the rate of economic development.

World Energy Consumption

During the last 100 years the rate of growth of the world's energy requirements has advanced at an even pace, with the exception of the two great wars and the depression. Concurrently, especially since the end of World War I, engineering developments have caused a marked improvement in fuel and power efficiency. In U. S. thermal electric power plants, to cite but one example, average efficiency has increased from less than 9% in 1920 to 24% in 1952. Because other improvements are known to have occurred in this country, it is certain that *effective* consumption of energy increased a good deal more rapidly than did consumption of commercial sources since World War I. Similar improvements occurred outside the United States, but probably on a lesser scale.

A number of countries (the mature industrial economies such as the United States and most of the Western European countries) today account for more than 60% of total energy consumption. Their rate of growth in energy consumption has been long and stable enough to be valid for estimating future requirements. For 1950–1975, the annual rate of increase in the United States will be approximately 3%. In the non-Soviet world, the increase will be about the same, with slightly less than 3% for Western Europe and slightly higher than this for the non-Soviet world exclusive of the United States and Western Europe. Together with other estimates, a reasonable rate seems to be 2–2½%.

(The United Nations reports that about 82% of the total *useful* energy was consumed in developed areas of the world, and it estimates that the rate of increase—barring war or depression—lies somewhere between 4 and 6%.)

A second group of countries includes those undergoing rapid industrialization, such as several Latin American nations (Brazil, Colombia, Mexico, and Venezuela), the Soviet Union, and certain countries in Eastern Europe. A 5% rate increase is estimated, but it is substantially higher in the U.S.S.R. These nations account for 20–25% of world energy consumption.

The third group includes underdeveloped areas—most of Asia and Africa and parts of Latin America. A very rough guess is a 4% rate increase. This group accounts for 15–20% of world energy consumption.

The world as a whole, then, will have a rate rise of about 3%, which means that requirements for primary energy materials must be doubled by 1975 and quadrupled by 2000. This indicates

that the rate of increase is not only likely to be high but very high. (See figure and table.) The principal governing factors will be the rate at which world output increases (population growth and increase in per capita productivity), extent of industrialization, changing efficiencies in energy production and use, and climate of the areas in which development takes place.

Energy Cost

During the last century, changes in the relative importance of energy materials, discovery of new sources, and improvements in transportation have reduced the geographical differences in energy prices. Differences remain, however, and they are still substantial.

Production and consumption of energy require capital layout and, for most underdeveloped areas, large amounts of foreign exchange. If there is no satisfactory rate of capital formation or access to foreign funds, not even low-cost energy will promote economic growth.

Any practicable increase in world energy requirements for the rest of this century can, if necessary, be met from present conventional sources—*at some price*. Price increases are inevitable with rapidly rising requirements, drain on oil reserves, and continuance of geographical differences. Oil has replaced coal to such an extent (because of increasing mining difficulties and other factors) that it has become decidedly the most important conventional fuel in international trade.

The cost of nuclear power will probably fall somewhere in the relatively wide range between low-cost energy from conventional sources at certain locations and high-cost energy at others. The tendency, in time, will be for it to move from the high side toward the low.

One of the most striking aspects of nuclear energy is the negligible cost of transporting the fuel. Therefore, those areas will benefit whose costs are now high because of distance from a fuel source. Naturally, the underdeveloped areas stand to benefit most from this, provided, of course, that capital and exchange difficulties are overcome.

Nuclear power may be more immediately profitable in high-energy-cost regions within capital-rich countries, such as the United States and Canada, because of capital availability and absence of exchange difficulties.

Outside these high-cost areas, use of nuclear power will probably depend on: (a) extent of increase in cost from conventional sources, and (b) extent of decline in cost of generating nuclear power. If, as seems likely, nuclear power undergoes developments similar to other new technologies, then lower costs are a certainty. Further, costs and availability of conventional sources may become serious handicaps. So pressing are these prospects that both the United States and the United Kingdom have begun nuclear power programs.

Even if the cost of nuclear power does not drop below conventional methods, expansion of nuclear facilities will make an important contribution because they will postpone an otherwise inevitable rise in the cost of energy from conventional sources. Nuclear power will weaken drastically the dependence of industrial expansion on particular sources of power. For undeveloped areas ready for economic expansion but lacking conventional supplies, ample quantities of nuclear power at lower costs can speed up their growth. For the world as a whole, a virtually limitless source of energy opens incredible vistas of economic expansion and higher standards of well-being.

Nuclear Power—Capital Outlay

Large-scale production of nuclear energy on a practical basis means large-scale capital outlay. The nuclear-energy industry does not, at present, seem adaptable to small-scale, inexpensive development. Investment costs can be reduced by a share-and-share-alike system among all nations concerned with nuclear development—in know-how, supply services, materials, and transport.

Like the United Kingdom (*390*), the United States (*476, 477*) has recognized the urgent need for a combined government-private industry nuclear program to meet the power requirements of the future. Under the aegis of the Atomic Energy Commission, two parallel programs in the USA are in progress—the Power Reactor Technology Program (Five-Year Program) and the Power Demonstration Reactor Program. Initiative in both programs rests largely with private industry. The capital outlay is enormous. But by 1960 the USA should have 750,000–1,000,000 kw of nuclear-power-plant capability representing an investment of $250–350,000,000. At the end of 25 years, it is entirely possible that the country may have a nuclear capacity alone that is equivalent to our present generating capacity of 100,000,000 kw. The investment would be about $20-billion in nuclear plants, but would

TABLE 1—Conventional vs Nuclear Power Plants (1955) (*477*)

Type of plant	*Capacity (kw)*	*Cost ($/kw capacity)*
Coal		
Large*	400,000–600,000	130–135
Small	11,000–50,000	150–165
Gas	medium to large	75–100
Hydro		
Large*	100,000–200,000	115–120
Small	10,000–50,000	125–170
Diesel		
Large*	10,000–15,000	135–138
Small	500–1,000	162–177
Nuclear	200,000	350
		250 (1965)

* Cost nearly constant for higher kw.

TABLE 2—Investment Costs (*477*)

Component	*Cost ($/kw capacity)*
Exploration, mining, ore concentration	1
Purification, reduction	1–3
Enrichment	40–80
Fuel-element fabrication, cold	5
Fuel-element fabrication, radioactive	15
Heavy-water manufacture	30–50
Zirconium manufacture	1–2
Nuclear reactor power plant	210–250
Fuel inventory	20–40
Containment	10–20
Fuel processing	30–60

TABLE 3—Reactor Operation for 4–5 Mill/Hr Power (*476*)*

Type of reactor	*Cost ($/kw capability)*	*Fuel*	*Fuel irradiation time (Mw-day/ton)*
Pressurized heavy-water	200	Natural U, cooled and moderated with D_2O	5,000–6,000
Water-cooled and moderated	200	1% U^{235}	10,000
Fast breeder† (conversion ratio 1.6)	270	Core: 10% Pu^{239} 90% U^{238} Blanket: 1% Pu^{239} 99% U^{238}	(20% Pu burnup in core per cycle)
Gaseous homogeneous† (conversion ratio 1.15)	200	Core: 3 gm U^{233} per liter D_2O Blanket: 1,000 gm ThO_2 per liter D_2O	(75% U^{233} burnup per cycle)
Sodium-cooled, graphite-moderated, regenerative (6–7 mills/kwh)	240	2%-enriched U	10,000

* These operating factors are based on $40/kg natural uranium, credit of $15–30/gm of fissionable material produced, 90% load factor, and annual fixed charges of 15% for plant and 4% for fuel inventory.

† Fuel-processing cost of $1/gm or less.

actually be much larger because of the need for ancillary installations.

Conditions and Assumptions

Initial costs will be high because the nuclear-energy industry is starting from scratch without any basis for determining costs except by comparison with conventional plants (see Table 1). No exact comparison is possible. Capital investment, the fixed charges on it, and the operating costs will set the price of power. Coal, for example, costs 35¢ per million Btu, the equivalent of $175/kw in investment, with 80% load factor and 14% annual fixed charges. Therefore, if nuclear fuel costs can be brought well down, a considerable margin exists for offsetting the higher investment costs of a nuclear-power-plant system.

In time, with increased efficiency and know-how because of intensive research and development, it seems fairly certain that the costs of nuclear energy will drop to the level of conventional power methods, and perhaps become even cheaper. One of the reasons is that the cost of first-generation plants is determined by construction at entirely new sites, with the site development cost charged to a single unit. Further, nuclear power plants will require specialized facilities that need not be duplicated for multiple units in the same area but that are initially charged to the first unit. Multiple reactor units will probably reduce charges for special facilities as much as 10–20%. The second generation of nuclear power plants, and certainly the third, will prove as cheap if not cheaper than conventional sources.

The magnitude of component investment costs is reflected in Table 2. The type of reactor on which these figures are based is assumed to be a heterogeneous thermal converter feeding low-enrichment uranium, producing plutonium for reuse as a fuel, and producing power with an over-all efficiency of about 30%, with a load factor of 70–80% and a reasonably long fuel-element lifetime. The figures given in the table are not strictly additive because not all systems will use the same materials and there will be some duplication. A rundown of costs for various reactors is given in Table 3.

Outlook for Savings

Capital-cost savings can be made in several ways:

▶Several units at one site and with the same design criteria.

▶Assembly-line instead of tailor-made production, for both nuclear and ancillary plants.

▶Technological improvements—design simplification, plant-component capabilities, increased efficiency, and cheaper materials and components (especially in chemical processing, waste disposal, and refabrication of fuel materials). Technology must be successful; failure in any direction will mean prohibitively high costs for the entire nuclear-power system.

▶Shift of responsibility from government to private industry.

Admittedly, determining the capital outlay and operating expenses of nuclear power on the basis of the foregoing conditions and assumptions is sketchy. However, knowledge in this field, based on experience, will increase a great deal in the coming decade.

Nuclear Industry—Regulation

The nontechnical problems of industrial utilization of atomic energy are encountered at international, national, local, and industrial levels. Some of the problems that must be considered at each or all of these levels are public health, safety of the atomic-industry worker—precise biological data are not yet available for these two—and trained manpower to operate the nuclear industry (*319*, *322*, *400*, *778*, *856*, *857*).

Public Health

From what is known about the biological effects of radiation, intensified radiation background could lead to somatic effects in the exposed population and genetic effects in later generations. It is the responsibility of public health groups to prevent the radiation background from rising to too high a value.

Without precise biological data, public health must rely on the experience of industry during the last ten years. It must not be accused of unreasonably hindering the development of nuclear power. To determine reasonable measures for protecting

public health, a program is needed. This might include:

1. Training public health personnel.

2. Disseminating pertinent information. An international coordinating body such as WHO could assemble information and submit it for scrutiny by expert groups.

3. Studying somatic effects of low-level radiation. Qualitative data must be obtained by work on animals. Human beings exposed to radiation can be studied according to groups, such as (a) patients submitted to radiation therapy, (b) workers in medical radiology, (c) workers in the nuclear energy industry, and (d) workers in uranium mining.

4. Studying radiation-induced genetic effects in humans. This includes obtaining qualitative information by studying mammals and the effects on offspring of exposed humans.

Case Studies of Reactor-Radiation Exposures (*617*, USSR)*

	Patient 1 (300 r)	*Patient 2 (450 r)*	*Comments*
1st period (primary general reaction)			
Duration	3 days	3 days	Loss of appetite, general weakness, dizziness, nausea and vomiting most pronounced 1st day
Temperature	Normal		
Arterial pressure	Slightly lowered	Slightly lowered	
Pulse	Satisfactory	Slightly increased	Treatment: stomach lavage; caloric diet rich in proteins and vitamins; vitamins B_1, B_2, C; penicillin up to 800,000 units/day; 200 ml of whole blood every 3–5 days
Hematologic effects	Leukocytosis, lymphopenia; profound depression of regenerative processes in bone marrow		
2nd period (latent period, apparent general good health)			
Duration	21 days	16 days	Weakness, dizziness, nausea passed away; appetite appeared. Slowly declining muscular tone. Lesion set in. Patient 1 started loss of hair on 13th day
Temperature	Normal		
Arterial pressure	Unstable	Higher	
Pulse	Unstable	Unstable	
Hematologic effects	Moderate leukopenia, lymphopenia; continued depression of regeneration, reduced number of blood-forming cells in marrow		Treatment: 12th–15th day penicillin increased 1½–2 times; streptomycin intravenously; 10.0 ml/day 10% calcium chloride; up to 1,000 mg ascorbic acid and vitamin K
Nervous system	Focal neurological symptoms; drastic reduction of motor chronaxies in later part of period; decrease of rheobase		
3rd period (marked manifestation of disease)			
Duration	20 days	40 days	General weakness, insomnia. Poor appetite. Patient 1: muscular tone abruptly declined; achilles and abdominal reflexes disappeared. Patient 2: rise in tendinous and periosteal reflexes with asymmetry in knee and achilles reflexes; abdominal reflexes weakened. Stool, urine normal. Gums loose, bled. Petechise in several skin areas Treatment: antibiotic treatment of oral cavity; hygienic treatment of skin; continued diet and transfusions; cardiatonic drugs and oxygeno-therapy as needed; blood-formation stimulants
Temperature	Up to 38° C	Up to 40° C	
Arterial pressure	90/70–105/70 Hg	100/40 Hg	
Pulse	74–92/min, rhythmic	100–110, rhythmic	
Hematologic effects	Acute leukopenia, moderate decline in hemoglobin and erythrocytes; appearance of regeneration in marrow		
Nervous system	Changes of general character, but not profound; tendency towards normalization shown by greater amplitude of chronaxy shifts and of reactions in cortex		
4th period (recovery)			
Improvement of general condition: lowering of temperature to normal; disappearance of hemorrhagic symptoms; gradual normalization of blood formation; in case of more severe case, disengagement of necrosis sections (tonsil)			
Exhibited long-time lability of pulse, skin vasomotor reactions, and arterial pressure			
Patient 1 discharged and given work precluding irradiation 3 months after appearance of disease; patient 2 returned to work after 4 months. Examination during 18 months indicates stable recovery			

* These exposures were the result of violating the operating rules for an experimental reactor.

Protecting the Worker

With the present status of knowledge, three approaches can be made to radiation protection: (1) the actuarial or statistical approach (determine the hazards from case histories), (2) provide protective devices and codes in such detail as to preclude the possibility of accident, and (3) the preventive approach.

The last approach, used by the U. S. Atomic Energy Commission, has enabled handling enormous quantities of radioactivity during the past 14 years in operations involving some 134,000 contractor personnel. Lost-time injuries per million man-hours in the U. S. are: electrical utilities, 11; nonferrous metals and products, 10; average for all industries, 8.2; miscellaneous manufacturing, 6.1; chemical, 5.5; AEC, 2.3; and communications, 1.8. It is borne out by the AEC figure that a vigorous employee safety program strongly supported by top management pays dividends.

The U. S. AEC spends about $100/yr/man on radiation protection. This is about 1% of the total operating expenses. To protect against radiation by an additional factor of 10 would cost 2% of operating expenses. This additional 1% would correspond to 20% of the total profits of a commercial enterprise, based on 5% net profit.

Regulating Nuclear Energy

The final pattern for radiation protection of the public and operators must be set by legislation. This pattern can be embodied in regulations giving specific details or spelling out main objectives to be met with the various radiation sources. For the flexibility needed by a rapidly developing art, the latter course is probably preferable.

Observations of Four Reactor-Radiation Exposure Cases (*478*, USA)*

	Patient 1 (male)	*Patient 2 (male)*	*Patient 3 (female)*	*Patient 4 (male)*
Dose (rem)	190	160	70	12
Symptoms	One episode of nausea and vomiting in several hours; disappeared without recurrence	None	Vomited on 3rd day, probably due to anxiety	None; on 15th day, small hemorrhage inferior to limbus of optic disk in right eye—disappeared
Urinary amino acid excretion (normal level—750 mg/day or 0.5 mg/ml)	During 1st 12-hr period: 1,556.1 mg. Peak: 5,500.0 mg during 6th day. Exceeded normal 5 months after irradiation	Exceeded normal on 3rd day. Peak: 4,422.0 mg on 6th day. Greater than normal 4 months after irradiation	Peak: 4.67 mg/ml on 8th day	During 1st 12-hr period: 637.6 mg. Peak: 4,830.0 mg on 6th day. Greater than normal 4 months after irradiation
	All excreted 14 or 15 amino acids per day compared to normal 5 or 6. This was apparent 12 hr after irradiation and as late as 5 months afterward			
Hematologic effects	Bone marrow studies on 10th day showed no abnormalities			
	Maximum depth of white cell depression reached 1 month after irradiation; total cell count changes of patients 3 and 4 were not sensitive indicator of exposure. All counts returned to normal 20 months after exposure in patient 1. Cytologic changes: greatly increased granularity of lymphocyte and leukocyte cytoplasm, frequent immature forms, many cells with karyorrhexis, inclusion bodies in lymphocyte cytoplasm. Minor cytologic abnormalities persist at 20 months			
Sperm studies	Patient 1: zero sperm cells per specimen at 10 months; normal at 20 months. Patient 4: minimum at 7 months (1/1,000 normal); normal at 14 months. Decrease and recovery of patients 1 and 4 was at same rate			
Frontal-lobe	All patients tested at 1–4 days after exposure; no objective evidence of impairment			
Eyes	All studied at yearly intervals; no lenticular opacities secondary to fast-neutron irradiation noted			

* These exposures occurred June 2, 1952, due to an accidental excursion of an ANL critical assembly. Assembly consisted of quasi-cylindrical metallic core immersed in a plastic tank filled with water.

In the United States, direct regulation will be mainly in the hands of the state governments. To provide some guidance, the National Committee on Radiation Protection established a subcommittee for considering the problem.

NCRP has completed a suggested form for a radiation-control act. One of the guiding principles has been to keep regulations as simple as possible. Where details and numerical quantities have to be included, they are limited to the factors that should be uniform over all the states. If this uniformity is maintained, then detail can be added without causing conflict among various states over basic requirements.

The NCRP recommends that the states adopt only those portions that apply to problems that may exist within the state and no more than can be adequately enforced. As experience grows, the coverage can be broadened. The following is a brief outline of the recommendations.

1. Scope and application of regulations.

2. Definitions. Only those needed for exact understanding of regulations. Over-use will encumber regulations.

3. Exemptions. Control not required for quantities so small that if taken internally, continuously or at one time, no harm would result. Appropriate levels are assigned to normally used radioactive materials.

4. Registration. May consist of writing letter to control agency or completing form supplied by agency.

5. Personnel and area monitoring. Indicates types of measurement that should be carried out. Surveys may be omitted where past experience indicates conditions are likely to remain safe.

6. Delegation of responsibility for safety in radiation installation. Hoped that state can shift direct responsibility to user. Thus, much less rigorous examination by state authorities will be required.

7. Storage of radioactive materials. Specifies storage so that escaping radiation will be below permissible levels.

8. Radioactive contamination control. Objective is to keep individual exposures below permissible amounts. Many practices are available; it is felt unwise to detail these factors.

9. Labeling. Provides uniformity in marking radiation work areas, sources, and containers.

10. Disposal of radioactive wastes. Regulations require that no materials be disposed of so that they concentrate at levels and in places harmful to human beings. Disposals must be with approval of state authorities; adequate records must be maintained. Where there is likelihood of exposing large population groups to radiation, levels are set at $\frac{1}{10}$ of those recommended for occupational exposure. Individual nonoccupational exposures may be averaged over one year.

11. Technical standards, guides, and general information. Numerical quantities should be subject to change as up-to-date information becomes available. Agency should be empowered to make changes. Data includes: (a) minimum quantities of individual isotopes subject to control; (b) maximum permissible dose from external sources of radiation (NCRP Handbook 59); (c) maximum permissible neutron fluxes (NCRP Handbook 55 and recommendations of ICRP); (d) relative biological efficiency of different types of radiation (recommendations of ICRP and NCRP); and (e) maximum permissible amount of radioisotopes in total body and maximum permissible concentrations of radioisotopes in air and water (NCRP Handbook 52).

REPORT from GENEVA

reactors

Design features and experience with power reactors, including the Borax tests—Power reactors under construction—Research reactors

Operational Power Reactors

Design features and operating experience with power reactors are summarized below. Figure 1 shows the 30-Mw reactor used in Russia's 5-Mw power station (see also NU, March '55, p. 70). Figures 2–4 show Argonne National Laboratory's prototype boiling-water power reactor (Borax-3).

Figure 5 shows the Borax-1 reactor, which was built to test transient behavior of boiling-water reactors; Fig. 6 shows a fuel element. Figure 7 shows the reactor shutting itself down safely by water expulsion after being made supercritical by 2.1% k_{eff}. This nondestructive ejection of water occurred when the minimum period of the excursion was 0.005 sec. After this reactor had served its purpose in the experimental study of over 200 short-period operations, it was destroyed as shown in Figs. 8–10.

To make Borax-1 destroy itself, a 4%-k_{eff} control rod was completely ejected from the core while the reactor water was at room temperature. The rod, which was completely out in ~0.2 sec., was only 80% removed when reactor power reached its peak; minimum period was 0.0026 sec. The excursion melted most of the fuel plates, burst the reactor tank, and sent most of the contents of the shield tank into the air. Total energy release was 135 Mw-sec; peak power was ~19 Bw; peak pressure was probably over 10,000 psi. Nuclear power release terminated at an early stage (0.003-sec heat flash was over before debris appeared above top of shield). Although spectacular, the explosion was moderate in intensity; most of the equipment outside the shield was either undamaged or repairable. At the control station ½ mile away, it sounded like 1–2 lb. of 40% dynamite on the bare ground at the same distance.

Most of the heavy debris landed near the shield pit. The ~1-ton control drive mechanism went 30 ft into the air and fell on the earth shield. Recognizable fuel plate pieces were found up to 200 ft from the reactor site. Most of the fuel plates had almost completely melted; some of the peripheral elements had only partly melted and parts of their fuel plates remained fastened to their side plates as in Fig. 11. Some melted fragments were found as spongy metallic globules as in Fig. 12; others had melted only inside as in Fig. 13. Practically all of the reactor's fuel was accounted for within 350 ft of the reactor, thus no large portion of the core left the site as airborne material.

The experiments with the Borax reactors prove that these reactors are inherently highly safe and indicate that boiling-water reactors can be designed to be safe from any practical reactivity accident.

The world's first electricity to be generated from nuclear energy was derived from heat developed by the Experimental Breeder Reactor in Figs. 14 and 15. The design is quite flexible both as to loading and operating conditions. It was designed when enriched uranium was not available in large amounts—so it has a small critical mass (48.2 kg U^{235}). Practical fast power reactors could be of somewhat similar design, but more stainless steel and sodium will be present for structure and cooling. This will necessitate a much greater dilution of fuel—and a degraded median energy of neutrons.

ATOMIC POWER STATION (APS-1). Russia, 30 Mw Heat, 5 Mw electricity, 5% enriched U fuel, graphite moderated, water cooled, $\phi_{avg} = 5 \times 10^{13}$, critical on 5/9/54, generated power on 6/27/54 (615).*

Description. Fuel alloy is contained in hollow SS-clad tubular elements in vertical SS-lined channels through stacked graphite bricks (whose T_{max} = 650–700° C). There are 128 channels; reactor is critical with fuel in 60; fuel charge is 550 kg U. Active core is 5.6 ft high × 4.9 ft diam. and is surrounded by graphite reflector. Reactor is hermetically encased in steel cylinder, which is filled with He or N_2 to prevent graphite oxidation; graphite has separate water cooling system. U burnout is 15%; enrichment decreases from 5 to 4.2% during operation. Pu production is only 0.32 because of low resonance capture. Shielding is provided on sides by 3.3 ft water and 9.8 ft concrete, and on top by extra graphite, steel cover, cast Fe plate, and extra concrete.

* References are to UN papers on p. 94.

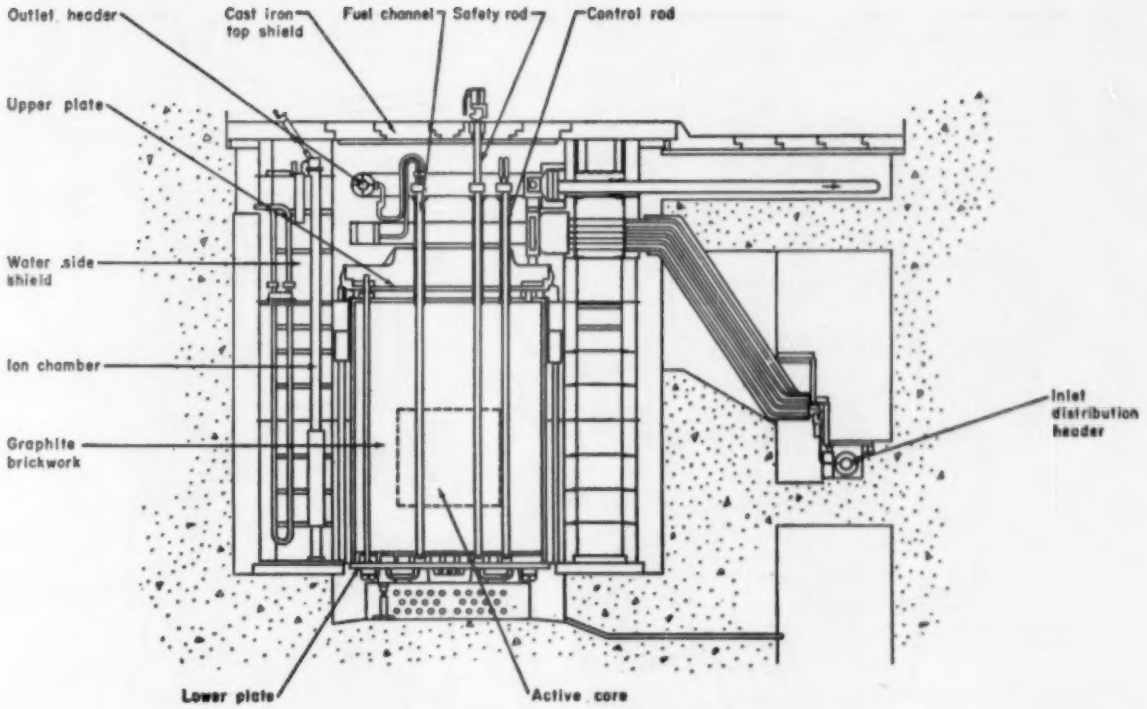


FIG. 1. Russia's pressurized-water graphite 30-Mw reactor used in 5-Mw power station

Heat. 1.5×10^9 cal/m³/hr heat is removed from U surface by distilled recirculated 1,470-psi water (flow and temp. are recorded for each channel) that flows from top inlet header, down axis of fuel tubes, then returns up by flowing over outer surface of fuel elements, at ~265° C it passes out to 3 of 4 pairs of counterflow steam generators (each consists of water heater, water evaporator, and superheater); water leaves steam generators at 190° C and goes to 3 of 4 main coolant pumps handling 300 tons/hr. Condensate pumped from turbine to steam generators is turned into 185-psi ~260° C steam at rate of 40 tons/hr. To hydraulically seal shafts of main pumps, makeup pumps feed water to seals of main pumps at ~8 psi above main system pressure. Storage batteries provide emergency power for afterheat cooling.

Control. 18 boron-carbide shim rods—6 are near center, 12 are at extremities of active core. They are moved vertically in special water-cooled channels (having separate cooling system) by wire ropes and servomotors. 4 automatic regulating rods (2 work at a time) in the reflector maintain power within 3%. Two safety rods have servomotors directly over top shield. There are 12 emergency shutdown signals. Most important ones: 20% over maximum power, too-rapid power increase, cut-off of main-pump power, certain changes in flow and temperature of water in fuel channels. At startup, sufficient excess reactivity is present for 2½ months of full power operation. Water flow is almost completely stopped to a given channel if it ruptures.

Irradiation facilities. Include: 6 curved channels to core boundary and reflector ($\phi = 2\text{–}8 \times 10^{12}$); 3 straight horizontal channels, 1 of which goes to core center ($\phi = 10^7\text{–}10^9$ at beam outlet); 1 thermal column ($\phi = 10^9$) with remotely controlled plugs so objects can be charged without shutdown.

Experience. Plant has produced $\sim 15 \times 10^6$ kwh. No fuel element failures; fuel channels easily removable. Cooling and automatic control equipment functioned without mishap. Fuel elements are replaced every 2 months, change takes 2 or 3 days. Startup from zero to rated power takes ~½ hr; automatic startup system is being developed. Moving peripheral elements to center extends U^{235} burnup to 20%. Have found only few pits and cracks in cooling system, although there are several thousand welds. Erosion of Cu gaskets and asbestos-graphite valve seals has caused some dry residue to build up. At temperature and pressure in cooling system, detonatable water-decomposition ions recombine at a rate such that recombiner could be shut down. Power from this station compares favorably in cost with that from small thermal plants, but considerably exceeds cost from large plants (10 kopeks/kwh = cost of electricity from Russian thermal power stations in 1953).

PROTOTYPE BOILING-WATER POWER REACTOR (BORAX-3). NRTS, Idaho (ANL project). 15 Mw heat, ~2 Mw electricity, ~90% enriched U, water moderated and cooled, 1955. Built for experience with operating power plant. Whole core assembly can be replaced easily (851).

Description. U-Al alloy is in Al-clad plates with 24 vertical plates in each box-type element, whose total dimensions (including top and bottom extensions) are 56½ × 3.875† × 3.828 in. Active portion of plate is 25.8 in.; additional ½ in. of Al is on each end. Plate centers are 0.324 in. apart; volume ratio of metal to water in core = 0.36; reactivity loss from 27 to 182° C = 0.8% k_{eff}. Fully loaded, core contains 87 fuel elements; 11.8 kg are required to cover criticality, temperature, operating Xe losses, and 2 kg for burnup. Each element costs $425 to fabricate; 137 gm U^{235} are in each central element; each peripheral element has 233 gm U^{235} so as to flatten power distribution and adjust void coefficient. Average element delivers 138 kw at 12 Mw total power. With 6.26-liter active section of each element, avg. power density is 22 kw/liter. Core is on framework that is supported from top of vessel (just below flat-lid closure). Vessel is ¾-in. SS, 15 ft 11¼ in. high and 52½ in. i.d.

Heat. ~1.1×10^8 cal/m³/hr heat is avg. heat flux (max. is 2½ × this) removed from 215° C surface of heated fuel plate by naturally circulating 300-psi water in core. Each cooling channel is 0.265 in. thick, 1.27 in. wide and 26.8 in. long. Water flows up through elements and steam formed collects above water surface 4 ft above core, leaves vessel through 6-in. pipe

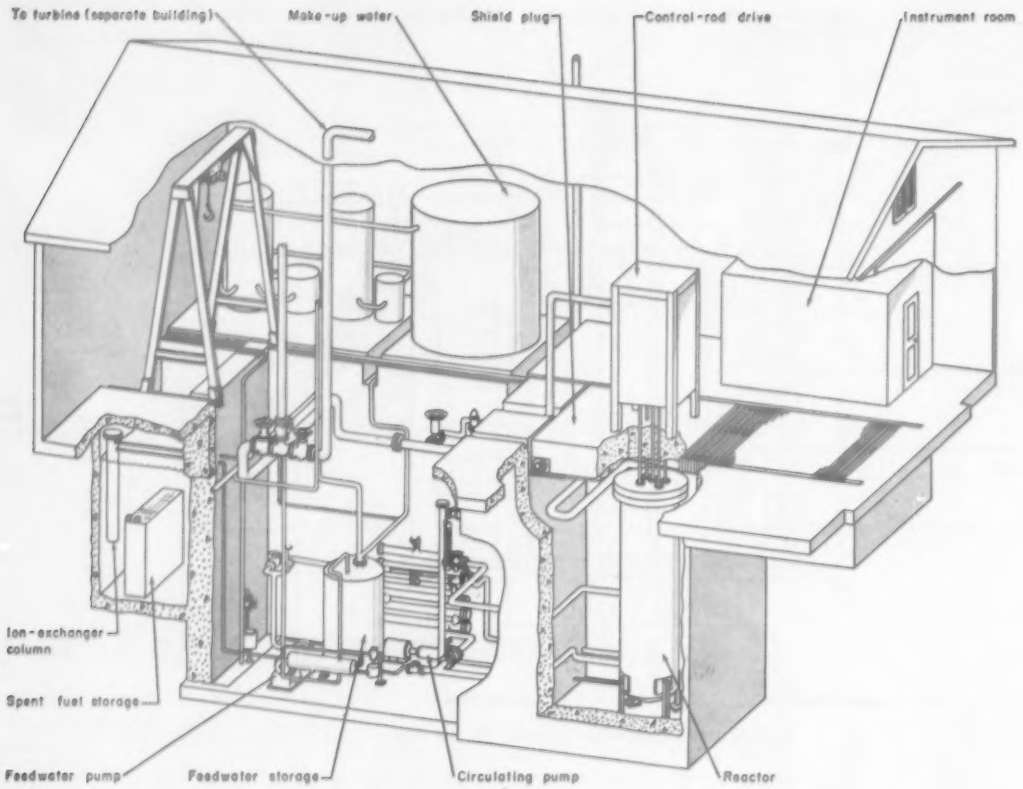


FIG. 2. US's 15-Mw prototype boiling-water power reactor (Borax-3) plant

and is vented to atmosphere or delivered directly to 3.5-Mw turbogenerator. Large steam volume in top of vessel is effective in raising steam quality. As water cools, it flows downward near vessel walls (where it serves as reflector); feedwater from condensate deionizer is pumped into the annular downcomer to promote natural circulation.

Control. 1 cross-shape and 4 blade-type control rods using Cd, B, and Hf as absorbers are driven vertically by drives mounted above concrete shield plug rolled over top of reactor. For each, steam cylinder and piston in which rod terminates balances lifting force due to steam pressure in core; steam for balancing cylinder comes from reactor vessel. Pneumatic piston exerting 800 lb drives rod. Rapid insertion is by switching air pressure to side of cylinder that moves rod down. Since facility also is used for transient testing of reactor cores, the control room is some distance from the reactor itself. Operating conditions are indicated by electrical means (including TV). For continuous operation, where burnup steadily reduces available reactivity, burnup plates incorporating boron are affixed to fuel elements.

Experience. Has operated under stable conditions at pressures from atmospheric to 300 psi. At 12 Mw and 300 psig (where power density = 22 kw/liter of core, which corresponds to 1.5% Δk in steam), short-period power fluctuations amount to 3.2%; these produced no detectable variations in reactor pressure or steam flow. $550,000 is total cost of facility; over 50% is in power conversion and cooling equipment. With 80% load factor and 20% efficiency, total electricity produced is 1.4×10^7 kwh/yr. With 15%/yr amortization, capital charges = 5.9 mills/kwh. If all 11.8 kg of U^{235} must be changed after 2 kg are burned, 1.8 fuel charges will be needed per year and consumption will be 3.6 kg/yr. To allow some radioactive decay of fuel, 1.5 loadings always will be committed to plant; with U^{235} assumed to cost $15/gm and 4% interest, inventory costs $10,600/yr, or 0.76 mill/kwh. The 3.6 kg burnup adds 3.85 mills/kwh. Fabrication of 1.8 fuel charges/yr at $425/element adds 4.7 mills/kwh. Processing at $5/gm (presently practical) adds 6.2 mills/kwh. A 12-man operating staff adds 8.6 mills/kwh, bringing total power cost to 30 mills/kwh. For this small a plant, capital and operation charges are fairly fixed, but considerable cuts are possible in cost of fabrication and processing.

BORAX-2. NRTS, Idaho (ANL project), 13 Mw heat, ~90% enriched U, water-moderated and cooled, 1954. Built to test transient performance of boiling-water power reactors (*481, 851*).

Description. Similar to Borax-3 (used same vessel) except that: core was considerably smaller, no turbogenerator was used, and fuel elements were narrower and somewhat different —similar to elements of Borax-1 except that there were only 10 fuel plates/element; plate centers 0.324 in. apart, volume ratio = 0.422; 93.4 or 157.3 gm U^{235}/element, reactivity loss from 27 to 182° C = 1.16% k_{eff}.

Experience. At 5.2 Mw and 300 psig (where power density = 22 kw/liter of core, which corresponds to 2.6% Δk in steam), short-period power fluctuations amount to ~15%; these produced no detectable variations in reactor pressure or steam flow.

BORAX-1. NRTS, Idaho (ANL project), ~90% enriched U, water moderated and cooled, 1953. Built to test transient performance of boiling-water power reactors (*481, 483, 851*).

Description. U-Al alloy in Al-clad plates 60-mils-thick with 18 vertical plates in each ~3-in.-square box-type element, which was not as wide as Borax-3's and was of different design; 138.6 gm U^{235}/element; plate centers 0.177 in. apart, volume ratio = 0.626; reactivity loss from 27 to 182° C =

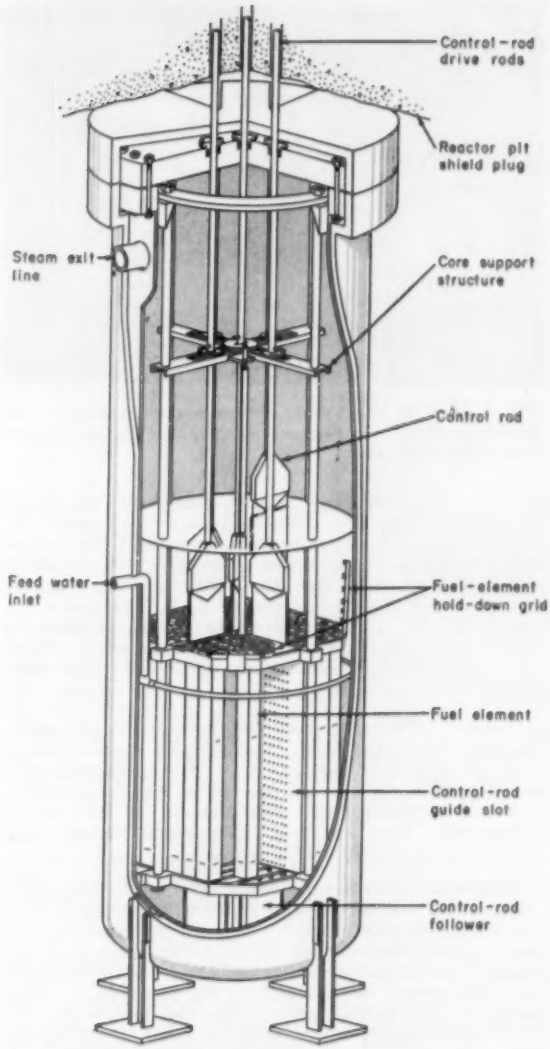


FIG. 3. Borax-3 reactor cutaway

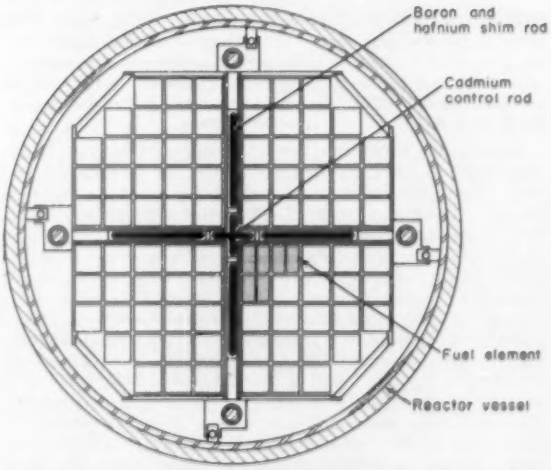


FIG. 4. Cross section through Borax-3 core, showing flat fuel plates

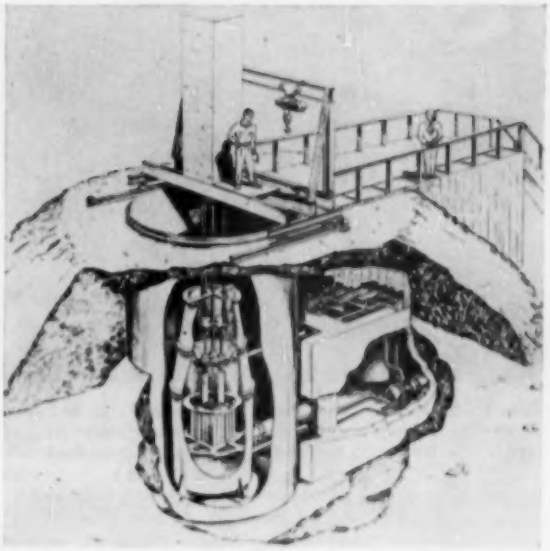


FIG. 5. Borax-1 reactor cutaway

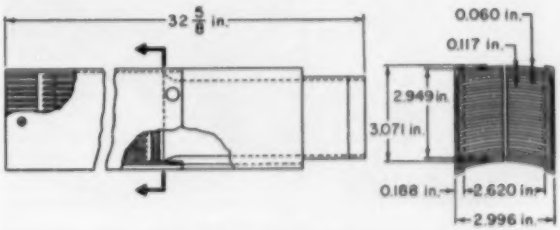


FIG. 6. Borax-1 fuel element used 18 plates each 60 mils thick; Borax-2 element had 10 plates of same thickness; both element types had external dimensions shown

FIG. 7. Borax-1 shutting itself down safely by water expulsion after being made supercritical by 2.1% k_{eff}

FIG. 8. When a 4%-k_{eff} control rod was 80% ejected from Borax-1's core there was a light flash as reactor power peaked near 1.9×10^{10} watts, then a dark gray column appeared

FIG. 11. Peripheral-fuel-element side plate with attached cluster of fuel plate fragments

FIG. 12. Globule of spongy Al-U from fuel element that evidently had been molten

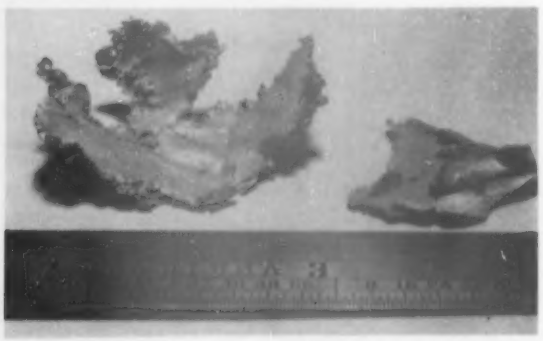


FIG. 13. Fuel-plate fragments that appeared to have been molten inside while outside remained solid

FIG. 9. Part of box housing control equipment can be seen starting to go up. Light-colored objects probably are Al-U fuel fragments burning in air

3.5% k_{eff}. Elements held in lower supporting grid and removable top cover grid. Reactor tank, of ½-in. steel, was 4 ft in diam. and ~13 ft high; it was in 10 ft diam. shield tank sunk partly into ground and with earth piled around for added shielding; shield tank filled with water only when reactor was shut down. Adjacent to shield tank was a concrete-walled pit with equipment for filling and emptying reactor and shield tanks and for preheating water in reactor tank; water level in reactor tank was 3–4½ ft over core. Facility located outdoors.

Control. 5 Cd elements operated by drives in rectangular housing above shield tank. Spring-loaded magnetic couplings connected mechanisms to rods. When released, center rod dropped out of core to apply excess reactivity used for experiments. Other 4 rods were dropped into core to stop experiments. Each rod took ~0.2 sec to traverse core. Control station was ½ mile from reactor.

Experience. After steam pressure (built up in forcing water rapidly from reactor during short-period experiments at atmospheric pressure) caused permanent deformation of fuel plates (max. temp. still did not melt fuel elements), it was decided that the reactor, which had fulfilled its experimental purposes, should be sacrificed in an experiment violent enough to melt the fuel plates. This was done on July 22, 1954. After explosive experiment, control-rod drive mechanism and most other equipment outside shield tank were decontaminated, reconditioned, and used on Borax-2.

EXPERIMENTAL BREEDER REACTOR (EBR-1). NRTS, Idaho (ANL project), 1.4 Mw heat, fast, ~90% enriched U, NaK cooled, fast $\phi_{max} = \sim 10^{16}$. Built to gain purely experimental experience with fast breeder reactor system. Produced >100 kw of electrical power in Dec. '51—this was world's first nuclear power (813, 814, 816).

Description. U^{235} is in slugs 4¼ in. long and 0.384 in. diam.; with 0.064-in.-thick SS jackets; 2 slugs are in each vertical rod, which is included in a close-packed array on 0.494-in. centers. Over and under fuel in rods are 0.384-in.-diam. U^{238} slugs that make blanket at ends of core; each rod has upper space for collecting fission gases and a top shielding section of SS that extends through part of reactor's top shield to serve as handle for removing rods; over-all length is 10 ft. Plates over and under blanket sections position rods in core; there are 217 spaces, but those not used for U^{235} rods contain blanket slugs. Core contains 52 kg U^{235}; critical mass is 48.2; reactivity loss from 38 to 200° C = 0.53% k_{eff}. U^{238} side blanket is in 2 sections; first part is separated from core by ⅛-in.-thick SS hexagon 7½ in. across flats. It consists of tight array of 138 U^{238} rods each $\frac{15}{16}$ in. diam. and 20¼ in. long inserted in 0.020-in.-thick SS tubes; they also have upper shielding and handling portions. This blanket part is in deep-drawn 15.9-in.-i.d. SS-347 can $\frac{5}{16}$ in. thick and 28 in.

FIG. 10. At height of explosion, large amount of debris in is air. Parts of reactor tank, controls, and core went 80 ft up. There was no dangerous fallout beyond a few hundred feet

long. Above this small-diameter section, the diameter of the reactor tank increases, this portion is filled mostly with steel to serve as a shield; the entire vessel is supported on the shoulder formed by the diameter change. Surrounding the small section is an outer tank of $\frac{1}{16}$-in. Inconel with ribs to maintain $\frac{1}{8}$-in. space (for insulation and part of main-vessel leak-detection system). The second blanket section is outside the reactor tank; it is of 84 keystone-shaped U^{238} bricks weighing ~100 lb each, jacketed in 0.020-in. SS and held in position in 12 stacks by an inner shell of $17\frac{9}{16}$-in.-i.d. $\frac{1}{16}$-in.-thick 28 Al and an outer shell of 31-in.-i.d. $\frac{3}{16}$-in.-thick 28 Al. Outside the external blanket and a 2-in. gap is a $\frac{1}{16}$-in. Al shell and an 18-in.-thick graphite reflector. Beyond this is an air-cooled 6-in. Fe thermal-neutron shield followed by 9 ft of concrete.

Heat. $>5.7 \times 10^3$ cal/m³/hr is avg. heat flux in core; 72% of heat generated by reactor is in core, 14% in inner blanket, and 14% in outer blanket, which is kept at 200° C by flow of 6,000 ft³/min air through 5 vertical holes in each of the 12 stacks. The air cooling of the blanket limits reactor power level. NaK at ~228° C enters reactor tank above inner blanket section, is distributed in header, flows through top plate in fluted section of rods and down between blanket rods to bottom of tank. From here it flows into the hexagonal separator and up among the fuel rods, through top fluted sections in holding plates, and at 316° C exits sidewards above inlet. Then it is cooled in heat exchanger by 215° C secondary NaK system and goes to a lower storage tank, from which it is pumped to a high tank and returns to the reactor. The secondary NaK heated to 310° C goes to steam generator and returns to a storage tank. The 405-psia 529° F steam is directed to a turbogenerator that makes electricity for plant operation.

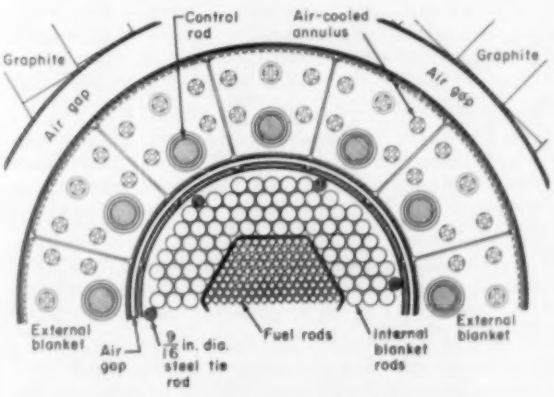


FIG. 14. EBR-1 horizontal cross section at midplane

Control. 12 SS-jacketed U^{238} control rods (1 in each outer-reflector stack of 2 in. diam.) move vertically to change neutron leakage from core; 8 can be pulled out of the blanket rapidly and serve as safety rods of 0.20%-k effectiveness. Together with a bottom safety (cylindrical block that can be driven out bottom by pneumatic force), these 8 rods can remove 0.27% k in 100 msec. Other 4 rods are regulating rods controlling 0.10% k. Whole outer blanket rests on lower shield on a hydraulically driven elevator; its top $4\frac{1}{2}$ in. of travel is mechanically controlled with 0.001-in. accuracy; through top $4\frac{1}{2}$-in. travel, 0.89% k is affected; complete dropping of outer blanket affects 8.2% k. Control-rod drives are under elevator. Loss of coolant has an effect on reactivity equivalent to removing 2 kg U^{235}.

Experience. Has operated reliably for $3\frac{1}{2}$ yr while being used for a number of experiments and generating 4×10^6 kwh heat, much of which went for generating plant electricity. Plant is inherently quite stable and largely self-regulating—very little operator effort is needed to maintain operating conditions. Instead of controlling reactor by load requirements, it is held at constant power and a pressure-regulating valve unloads excess steam to condenser. Plant efficiency is only 17% because of small size of plant and part-load operation. One method of determining conversion ratio gave a value of 1.00 ± 0.04; another method gave 1.01 ± 0.05. Strong circumstantial evidence indicates considerable neutron leakage out of blanket, a more efficient blanket probably would give a ratio of ~1.3. To replace the outer-blanket bricks, elevator lowers blanket, handling dolly lifts blanket from shield plug and takes it to shielded room, where manipulators disassemble it.

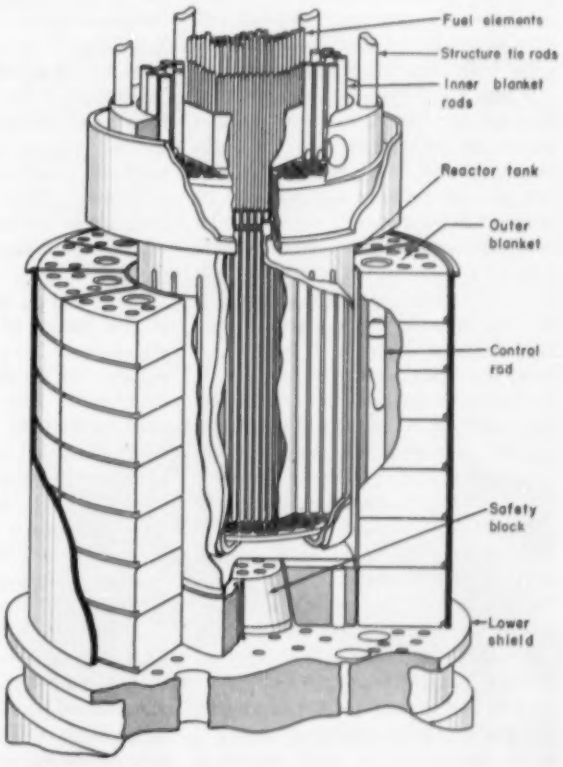


FIG. 15. EBR-1 cutaway

FIG. 16 Pressurized Water Reactor and its primary coolant system are in steel vessels sized to contain highest pressure that might be created by system rupture

FIG. 17. PWR core cutaway, showing structure and size

Power Reactor Designs

Design features of power reactors under construction or planned are described in this section. Figures 16–22 show the arrangement of plant and core for the Pressurized Water Reactor being built near Shippingport, Pa.

Russia's 100-Mw atomic power station will use two APS-1-type reactors producing 200 Mw heat each. Planned for completion next year, it will have 500 tons of graphite and will use 20 tons of 2.5%-enriched U per year. To boost the conversion ratio from APS-1's 0.32 to a value of 0.5, zirconium will be used instead of stainless steel in the core structure. On a site of only 12½ acres and with two or three times fewer operating personnel than a coal-fired plant of equal size, power produced will cost 10–20 kopeks/kwh (*615*). Factors in the design of a Russian-proposed boiling homogeneous power reactor also were presented (*624*).

Figure 23 shows the core of the Homogeneous Reactor Test being built at Oak Ridge National Laboratory. D_2O in blanket vessel will serve as reflector during initial tests; thorium-oxide D_2O suspension will be used as blanket later; finally, U^{233} may be used as fuel to evaluate U^{233}-Th cycle. The stability of homogeneous reactors, as demonstrated in power-transient tests conducted using the HRE, make it possible for the HRT to operate safely without control rods (*498*). HRE generated 150 kw of electricity on February 24, 1953 (*816*).

An ORNL preliminary design for a $20–25-million 100-Mw homogeneous reactor plant has a Zr 5-ft spherical core vessel containing U salt dissolved in D_2O that is heated to 300° C to generate steam in an external heat exchanger. The 2,000-psi D_2O-thorium oxide blanket is contained in an 11-ft-diam. annular pressure vessel (*496*).

Experiments are being conducted at Los Alamos to demonstrate operating characteristics of low-cost reactors suitable for generation of 20–25 Mw electricity. Two reactors are being used for the tests—one develops 1 Mw heat; the other yields 2 Mw. Both are fueled by phosphoric acid solutions of enriched U and have coil-shape heat exchangers inside the reactor pressure vessel. Gold plating reduces corrosion of structural materials contacting the fuel solution. Fuel temperatures of 350–450° C eliminate need for devices to recombine the radiolytic gases (*500*).

A 250-kw reactor to be built in the Netherlands will be used to determine characteristics of an intermediate-enriched UO_2-*wet*-suspension (D_2O) system with reflector of H_2O and BeO. It will operate at 250° C and have a neutron flux of 10^{13} (*936*). After consideration of three types of power reactors (homogeneous, heterogeneous, and quasi-heterogeneous) using *dry* suspensions of UO_2 with gas as the medium, it is concluded that the heterogeneous arrangement is most promising because it: can use natural U, can produce high-quality steam without having a pressurized reactor, and has a high conversion ratio—and might breed using thorium oxide (*938*).

Canada's Nuclear Power Demonstration Reactor is shown in Fig. 24. Its ~$11-million cost (which does not include development or land costs) will be covered by: ~$8-million from Atomic Energy of Canada Ltd. (which will provide nuclear data, fuel, moderator, and processing, and will be responsible for nuclear performance of the plant), $2-million from Canadian General Electric Co. (which will do detail design, engineering, and construction), and by ~$1-million from the Hydro-Electric Power Commission of Ontario (which will design and pay for turbogenerator and transmission equipment and operate the plant). The Commission will buy steam from AECL and will be reimbursed for reactor operation.

PRESSURIZED WATER REACTOR (PWR). Shippingport, Pa. (Westinghouse-Duquesne project). >230 Mw heat, >60 Mw electricity, highly enriched U and natural U fuel, pressurized-water cooled and moderated. Being built to gain experience with full-scale nuclear power plant; operate in 1957. Including building and turbogenerator, cost = $37.5-million (601, 604, 815).

Description. Two types of fuel units known as "seed" and "blanket" assemblies are arranged to form an active core 6 ft high with a 6 ft mean diameter. Each seed subassembly consists of a number of parallel vertical 0.080-in.-thick plates welded together to form a box ~3 in. on a side. Each plate is of enriched U-Zr alloy clad with Zircaloy-2; coolant channels are 0.080 in. wide. 4 box units are welded together with spacers to form a central cross-shape space for a cross-shape control rod. The 24 seed assemblies contain 52 kg highly enriched U^{235}; they are arranged in a 1-assembly-thick square annular array with 6 on a side. Area inside and outside annulus is filled with total of 12 tons of U in natural-U blanket subassemblies. Each of these is a stack of 7 bundles of 100 rods each 10 in. long. Bundles are fastened together mechanically; rods are welded on 0.468-in. centers to end plates in which holes are drilled to permit water flow between rods. Each rod contains UO_2 pellets in 0.413-in. o.d. Zircaloy-2 tube with 0.028-in. wall and welded end plugs. All the 119-in.-long seed and blanket subassemblies have same cross-sectional shape and are fitted at each end with transition sections that terminate in circular tube pieces. These fit in holes drilled in a SS support plate at the bottom and in a SS top grid both of which are parts of ~8-ft-o.d. 13-ft-high SS barrel cage supported from a ledge of reactor pressure vessel. This is a 9-ft-i.d. cylinder with a nominally 8½-in. wall of carbon-steel plates and forgings with 0.25-in. SS cladding. It is 33 ft high over-all, has 6-in.-thick hemispherical bottom and standard flanged ~10-in.-thick top modified to use a gasket or seal weld for final closure after bolting with 42 6-in. studs. Entire core can be removed through head opening of 109 in. i.d. Top head is pierced by control-rod drive rods and by small ports through which core subassemblies can be removed individually. Vessel is surrounded by annular tank providing 3 ft thickness of water. Concrete provides additional compartmental and exterior shielding. Plant container is in four units connected by large tubular ducts. Reactor vessel is in 38-ft spherical section with 18-ft cylindrical dome to accomodate control-rod drives, which extend 11 ft over core vessel; two main coolant loops are in each of two 50-ft-diam. cylindrical sections. Fourth cylinder, 50 ft in diam. and 147 ft long, contains auxiliary systems. Each container is in a concrete compartment partially below ground level.

Heat. Maximum heat flux in seed assemblies = 382,000 Btu/hr/ft² ≈ 10^6 cal/m²/hr, (avg. = 112,000 Btu/hr/ft²); in blanket max. ht. flux = 240,000 (avg. = 65,000). Peak power density is 277 watts/cm³ in seed and 120 in blanket. 90% of the 45,000 gpm of high-purity water at 2,000 psi and 508° F that flows into bottom of core vessel from 3 of 4 loops (only 3 required for 60-Mw generation) flows up through orifices between fuel plates and rods at 10–20 ft/sec; 10% bypasses core to cool vessel walls and thermal shield. After water at 542° F leaves core it goes through 18-in. pipes at 30 ft/sec to 3 of 4 loops, in each of which it flows through two 16-in. gate-type isolation valves in series, a heat-exchanger section of a steam generator, centrifugal single-stage canned motor pump, and back through inlet isolation valve to reactor vessel. Dry and saturated 600-psia steam from steam separator goes to 100-Mw turbogenerator outdoors. Plant can be brought from cold shutdown to operating range in 3½ hr, as compared to ~5½ hr with conventional plant of like size. At start of reactor life, ~60% of power is produced in blanket and 40% in seed. Most of power comes from fissioning of U^{235}, but ~8% comes from fast fissioning of U^{238}, and a substantial fraction comes from the fission of Pu. Latter effect increases with life of reactor. At start of reactor life, 0.8 atoms of Pu are formed per atom U^{235} fissioned. Seed assemblies must be replaced at least once to get maxi-

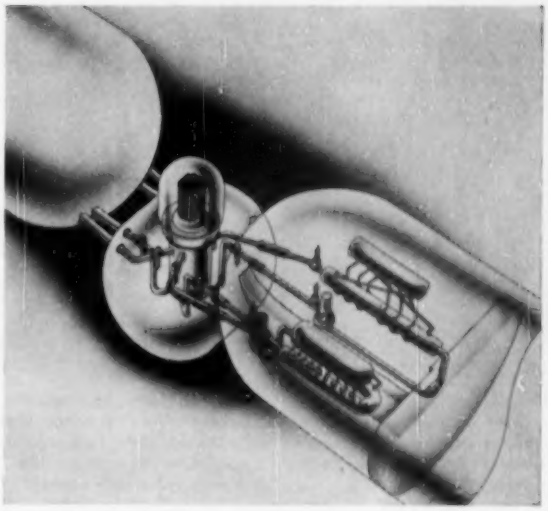


FIG. 18. PWR reactor is in spherical vessel. Two of the main coolant loops are in each of the two adjacent cylinders. Different types of heat exchangers (U-tube and straight-tube types) are used for design and performance evaluation

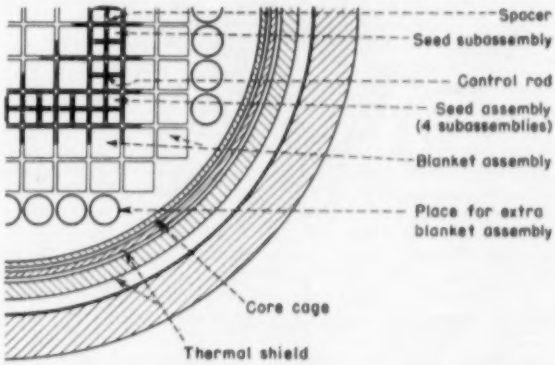


FIG. 19. Cutaway of part of PWR core shows location of control rod and 5½-in.-square seed and blanket assemblies

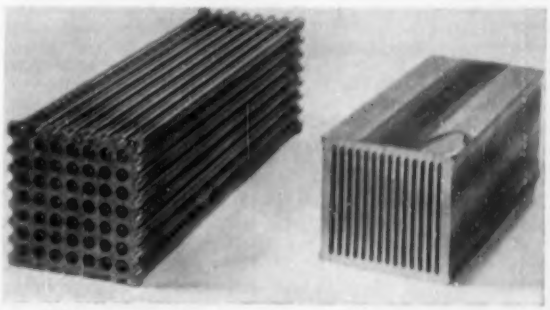


FIG. 20. Typical shapes of PWR fuel elements (actual elements contain more plates and rods). Blanket-element construction is shown at left, seed element configuration is at right

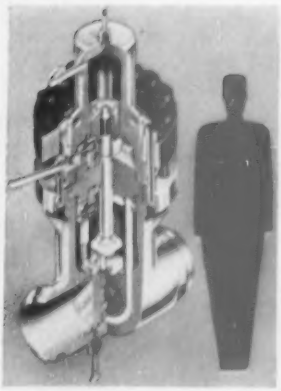


FIG. 21. Each of PWR's four coolant circuits has a canned-motor pump run by a single-winding 1,200-kw-capacity motor

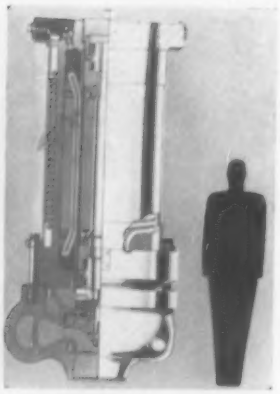


FIG. 22. Two hydraulically operated main stop valves are used in each of the four coolant loops. One shown is parallel-disk gate valve with piston

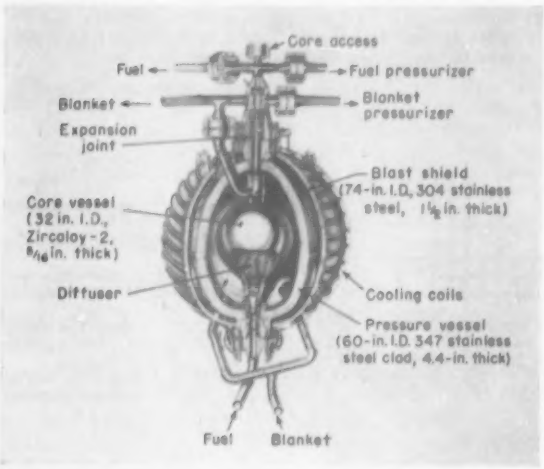


FIG. 23. HRT reactor vessel assembly

mum life out of the uranium oxide blanket assemblies.

Control. A cross-shape control rod is in cross-shape space between boxes in each of the 24 seed elements; homogeneous rods are of crystal-bar Hf, they move at 20 in./min and scram at >200 in./min. Negative temperature coefficient $\geq 2 \times 10^{-4}$ $\Delta k/°$ F. To increase loading of seed assemblies so as to give increased life, burnable poisons are added to fuel plates.

CALDER HALL. Cumberland, England; 2 reactors producing total of <92-Mw electricity; thermal, natural U, graphite moderated, pressurized-CO_2 cooled, 1956. Design dictated by requirements of safety in densely populated country. Low specific ratings are balanced by economy and simplicity of materials of construction (406).

Description. Fuel rods fit in vertical channels in graphite moderator within 40-ft-diameter 60-ft-high 2-in-thick welded steel pressure vessel that was fabricated on site. Weight of graphite (>1,000 tons) is taken on 4-in. steel base plate pierced for fuel channels and supported on diagrid steel-girder lattice spanning an I-section ring girder, which is supported by brackets through pressure-vessel wall onto 10 inverted A-frames that rest on bottom thermal shield. Fuel replacement is via charge tubes piercing pressure dome and concrete shield. Thermal shield to protect concrete is 6-in. mild-steel plate; octagonal-shaped biological shield is concrete several feet thick. Each reactor is in separate building.

Heat. Pressurized CO_2 enters core through manifold at bottom, flows up around fuel rods, and leaves side of vessel below its top head through 4.5-ft-diam. mild-steel ducts, and goes to 4 heat exchangers at corners of building. To form steam, CO_2 gives up heat as it flows down these 70-ft-high by ~18-ft-diam. vessels; then 4 main 2,000-hp blowers return CO_2 to inlet manifold. To prevent buildup of graphite dust and iron oxide in system, ~2% of mass flow through each blower is bypassed from downstream side, filtered, and reintroduced upstream. Steam goes to generator house (between the 2 reactor buildings) where it turns total of four 23-Mw dual-pressure tubogenerators at 11 kv. 20 tons of CO_2 needed to fill reactor and coolant circuits. Induced-draft cooling air flows between thermal and biological shields and is exhausted 200 ft above ground via 2 stacks atop reactor building.

Control. Control and safety rods, of stainless-steel tube lined with boron-steel fillers, are suspended through the reactor roof.

DOUNREAY, Scotland, fast breeder, ~60 Mw heat, unspecified U^{235}-U^{238}-Pu^{239} fuel and breeding system, Na cooled. Built flexibly for design experiments (405).

Description. Core of reactor, ~2 ft diameter × ~2 ft high, is in stainless-steel pot through which primary Na or NaK coolant circulates. Pot is surrounded by 4-ft-thick borated-graphite neutron shield to protect secondary coolant from activity-inducing neutrons. Graphite is heated to reduce radiation damage. Annular area containing primary-secondary heat exchangers is in turn surrounded by 5-ft-thick, 45-ft-high, 90-ft-o.d. concrete biological shield, whose bottom section supports the steelwork that holds pot and core. Roof slab, with removable 20-ft-diameter center hole, is supported on a ring of vertical columns that pass through heat-exchanger space. Entire reactor is enclosed in 135-ft-diam. 1-in.-thick steel sphere designed to contain inside pressure of 18 psi. There is ample space for large crane in the upper hemisphere.

Heat. Coolant loop, of stainless steel throughout, has no valves or dump tanks. Primary circuit has 24 parallel loops, each with 2 separate heat exchangers and an electromagnetic pump. Core is fed by 20 loops from common header; blanket is cooled by other 4. Secondary loop, Na or NaK, carries heat to boiler for steam generation.

HOMOGENEOUS REACTOR TEST (HRT). Oak Ridge, fast breeder, 5–10-Mw heat, UO_2SO_4-in-D_2O core (U is 90% enriched), D_2O blanket, 1956. Prototype of full-scale power plant (496, 498, 816).

Description. Inlet to diffuser is 3/8-in. plate in form of 30-deg cone followed by 90-deg cone; 9 perforated 1/8-in.-thick plates distribute the fluid entering from the 3½-in. inlet pipe. Core tank can stand internal pressure >900 psi, external >400 psi, but pressure differences greater than 200 psi will not be permitted. Pressure vessel that contains the 2,000-psi working pressure is ASTM A212-52at carbon steel clad with 0.4 in. of 347 stainless and can stand 18,000 psi plus thermal stress of 10-Mw operation. Nevertheless, 1½-in.-thick blast shield is provided in case radiation damage embrittles vessel in spite of annealing effect of 300° C ambient. Containment tank is 54 × 30.5 × 25 ft deep and is of welded ¾-in. steel plate reinforced to hold 30 psi (corresponding to instantaneous release of entire contents of reactor system). To decrease thermal-neutron activation of equipment inside tank, reactor vessel is surrounded by 1-2-ft-thick mixture of barytes sand, boron-containing Colemanite, and water (bulk density = 3 gm/cm³) in igloo-shaped steel tank. Shielding roof, high-density concrete 5-ft thick, is at ground level; reactor is in pit. Roof is two layers of removable slabs with completely welded steel sheet between the layers and extends across the top of the pit to form a gas-tight seal. Wall between reactor pit and control room is 5½ ft of barytes gravel and water between two ½-in. steel plates. Entire reactor pit can be flooded for maintenance. Freezer jackets around pipes

at disconnect joints freeze heavy-water ice plugs in line to prevent leakage of shield-flooding water into reactor piping during disassembly.

Heat. At 5 Mw, fuel is pumped into bottom of core at 256° C and emerges at 300° C; specific power is 17 kw/l. Then fluid is heated electrically to 335° C by pressurizer to produce 2,000-psi steam. Fuel flows past pressurizer to gas separator where directional vanes separate off radiolytic and fission-product gases (0.96 ft^3/sec at STP) which are bled off to separate low-pressure system that recombines D_2 and O_2 on catalyst; radioactive fission-product gases are adsorbed on activated carbon pending decay and discharge from 100-ft stack. Fuel solution continues on to U-tube heat exchanger where it is cooled to 256° C by transferring heat to boiler feed water surrounding the tube bundle. The 244° C 520-psi steam is bled partially to a 312-kva turbogenerator and partially to an air-cooled steam condenser. The UO_2SO_4 solution is next returned by a 400-gpm canned-rotor pump to the reactor vessel. Blanket fluid follows an identical cycle at 230 gpm. All this equipment is within the reactor shield pit.

Control. There are no control or safety rods because of inherent stability. Temperature control is by regulating concentration of fuel in the core; power level is set by rate of heat removal from the reactor. Large negative temperature coefficient of reactivity, $-2 \times 10^{-3} \Delta k/k/°$ C at 280° C, protects against large increases in reactivity. Safety of operation was demonstrated in the Homogeneous Reactor Experiment. Large temperature coefficient of reactivity means that reactor cannot generate more heat than is being removed; thus average core temperature will always be 280° C for a fuel concentration of 9.6 gm U^{235}/kg D_2O, although inlet and outlet temperatures will vary in proportion to power extraction rate.

SODIUM REACTOR EXPERIMENT (SRE). (North American Aviation Inc. project) Santa Susana, California, 20 Mw heat, 7.5 Mw electricity potentially, thermal, 2.8%-enriched U, graphite moderated, Na cooled, 1956. Explore Na-graphite technology (499).

Description. Reactor is located below grade with upper surface of top shield at floor level. Entire core is located in a 1½-in.-thick 304-stainless-steel vessel 19 ft deep and 11 ft in diameter. Fuel elements (~37 for 20-Mw design) are clusters of 6 rods close-packed around a seventh central rod. Each rod is a 6-ft-high column of 6-in.-long ¾-in.-diameter U slugs in a 0.010-in. 304-stainless jacket; rods are bonded to jackets by NaK in an 0.010-in. annulus. The stainless-steel jackets are welded closed at each end with a stainless plug. Rods are wrapped with 0.092-in. stainless wire with ~10-in. pitch to avoid rod contact and to mix coolant. For expansion, there are a few inches above the top slug filled with He gas that is at atmospheric pressure at room temperature. The seven rods forming each cluster are retained at their ends; at the top is a handling and supporting fixture, while individual expansion is permitted by bottom fixture that also locates the elements on the bottom grid. The moderator consists of graphite prisms of hexagonal-cross-section in 0.035-in. Zr sheet cans with 0.10-in.-thick ends. The distance across the flats of each can assembly is slightly under the 11-in. spacing of the triangular moderator lattice to leave 0.170-in. average gap for can-cooling Na flow. Sides are dimpled to guard against closing off Na flow. Moderator cans in the core have a 2.80-in.-i.d. 0.035-in.-wall axial Zr tube welded to top and bottom ends. These tubes ordinarily contain fuel elements. To equalize pressure there is a ¼-in. Zr vent tube extending from the sealed-off bottom of each moderator can out through the top into the He-filled space above the core. There is a collector cup at the bottom of the vent tube to collect Na that may diffuse in during the lifetime of the unit. To provide space for control rods and experiment holes, certain trios of moderator cans are shaped along their adjacent corners to provide a 3¼-in.-diameter channel. Thermal shield is a 19-ft-high stack of seven 5½-in.-thick low-carbon-steel rings that rests on the flat bottom of the 12½-ft-diameter 19-ft-high outer tank. Outer tank also supports core tank and provides emergency reservoir for Na in event of core-tank leakage. Between outer tank and ¼-in.-thick cavity liner there is a 12-in. annulus filled with Superex block insulation maintained at 3-psig of N_2 during operation. There follows a 3-ft.-thick concrete biological shield. Top shielding is a heat and thermal shield followed by 3.6-density magnetite-loaded concrete 6-ft thick. There is a gas-tight 60-ton 15-ft-o.d. 11-ft-i.d. ring shield encased in type-405 stainless steel. Within this is a 75-ton rotatable shield similarly encased. There are 81 small stepped gas-tight plugs extending through this shield and 3 larger ones, two 40 in. in diameter and one 20 in. Any graphite assembly within the core can be removed via one of the 3 larger holes by rotating the shield. To do this, one melts low-melting-point Cerrobend in tongue-and-groove seal at outer edge and raises and rotates shield with building crane;

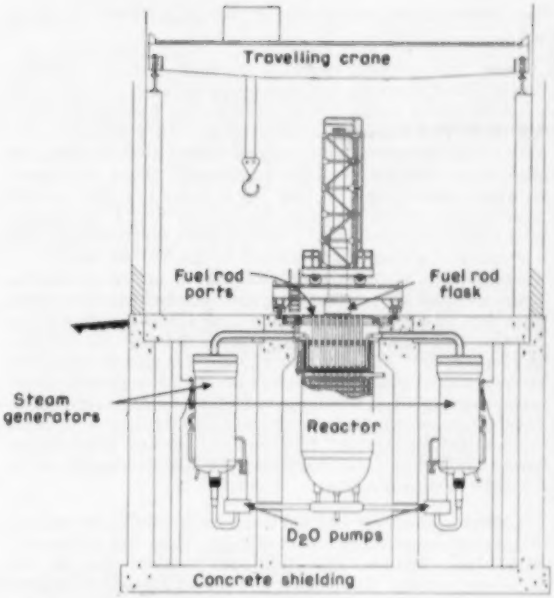


FIG. 24. Canada's NPD reactor

PWR Mock-ups and Tests

Here are some of the tests used to establish design information or to confirm adequacy of reactor components for PWR. They indicate the magnitude of work involved.

Vibrational characteristics of fuel and control elements.

Thermal-distortion characteristics of control elements.

Life and wear tests in hot water to establish reliable materials combinations for control mechanisms.

Strength and ductility tests on formed and welded components of fuel elements and subassemblies.

Crevice and fretting corrosion of mechanical fasteners.

Warping tendencies of fuel elements under severe temperature variations.

In-pile tests to determine adequacy of fuel-element shape and grouping under irradiation.

Flow distribution through core, to thermal shields, and to control elements.

Testing of core instrumentation: mechanical, thermal, and hydraulic.

Buckling loads of control-element drive shafts under various conditions of guiding and alignment.

Deflections under load of core-support and reactor-vessel head to learn effect on control channel alignments.

Natural-convection flow through core to learn adequacy of core cooling when no power is available.

fuel elements are raised into a 35-ft-high 75-ton handling coffin.

Heat. It is planned to attain maximum U metal temperature of 1,200° F in each fuel element by adjusting Na flow and selecting orifice plates for fuel elements. Central elements will carry 17,500 lb Na/hr at 5 ft/sec with 2.5-psi pressure drop across element and 1.5 across orifice plate to dissipate the 650 kw (340 kw/kg of U^{235}) produced. Flow in element requiring least cooling will be ~3 ft/sec. For 500° F inlet temperature, outlet temperatures will be 908° F for central tube and maximum of 986° F for a tube in the lowest average neutron flux, giving a mixed mean in the Na pool of 960° F. There are two circuits to provide Na—main and auxiliary. Main loop, of 6-in.-diameter type-304 schedule-40 stainless pipe, has 13.5-ft/sec Na velocity and 17-psi pressure drop. Auxiliary, of 2-in. diameter pipe, has 5.5-ft/sec Na velocity and 7-psi drop. All valves use frozen-Na stem seal topped by He gas seal to avoid oxidation. An extensive system circulates toluene to maintain frozen-Na seals. Intermediate U-type heat exchangers transfer 985 Btu/hr/ft²/° F in main system, 800 Btu/hr/ft²/° F in auxiliary system, to secondary loops that feed air-blast exchangers for final dissipation of the reactor's heat production.

Control. There are four control rods located at the corners of a square centered on the core axis. Each rod moves in a 23.5-ft-long 0.049-in.-wall type-304 stainless thimble for isolation from the Na coolant; however, a 16-psig He atmosphere is maintained within for thermal contact. Rods are made of a series of eighteen 2.5-in.-o.d. 1.75-in.-i.d. 4-in.-long B-Ni alloy rings with 2% by weight B. Arrangement permits about 10% control in reactivity. Control-rod drive is by ball-nut screw arrangement. Two of the rods have 0.24-ft/min shim drives while the other two have an additional 3-ft/min regulating drive with ±6-in. travel. There are also 4 safety controls with 10% reactivity control in similar thimbles. Fourteen B-Ni rings, $2\frac{5}{16}$ o.d. 2 in. i.d. and 4 in. long, are used in this case. Scramming is accomplished by unlatching with solenoid to permit free fall until 24 in. from the bottom of the thimble; here, piston arrangement slows it.

NUCLEAR POWER DEMONSTRATION (NPD). Des Joachims, Ontario, Canada, 10–20 Mw electricity, natural U (possibly Pu-enriched), D_2O moderated, D_2O cooled, 1958. Tentative design for working prototype. Cost ≥$11-million.

Description. Fuel is in 200 Zr-sheathed rods. Core is ~10 ft in diameter × 10 ft high. Al tubes wrapped around outside of core and containing H_2O may be used as a reflector. D_2O moderator, which is outside coolant tubes around fuel elements, so that it is not heated appreciably, is circulated through heat exchangers to keep its temperature at 60–260° F. Fuel rods supported and loaded from top. Entire reactor will be below floor level.

Heat. D_2O coolant, at 500° F when it leaves reactor, is kept liquid by >1,200-psi pressure. It gives up its heat to H_2O in two heat exchangers to make steam that goes to turbogenerator.

Control. There may be no control rods, control being achieved by varying quantity of D_2O moderator in core with level pump balanced against leak rate down tube. Pressure differential between He atmospheres above and below moderator supports it. D_2O moderator can be dumped quickly by removing He from bottom section.

BOILING HEAVY-WATER REACTOR. Norway, 10–20 Mw heat, slightly enriched U, D_2O moderated, 1958. Preliminary design for prototype (879).

Original design was a natural-U pressurized-D_2O reactor in which D_2O-cooled fuel elements would be thermally insulated to permit D_2O moderator to remain cool (this being the only way to assure criticality with unenriched fuel.) However, the prospect of obtaining enriched fuel prompted switch to this boiling-water design.

Description. Reactor is to be located below floor level of room blasted in granite mountain; access to room is through 60-ft tunnel. Fuel elements are 1 in. o.d., 0.4 in. i.d. and 4 in. long and sheathed in $\frac{1}{8}$-in. Al-alloy tubes. There are 373 elements, set in a 5-in. lattice provided by holes in 7-in.-thick steel base plate. Total length of U column is ~100 in.; total weight of fuel is ~9 tons. Total weight of D_2O moderator is ~15 tons. Operating pressure, about 400 psi, is contained by 11-ft.-i.d. 37-ton tank. Shielding at top is 7-in. steel in two pieces to absorb bulk of gamma-ray heat, followed by 16-in. 18-ton tank lid with mounts for supporting unfueled top ends of fuel-rod assemblies. Then, there are two 8-in. steel plates, lower one of which supports coffin when fuel is unloaded. Thermal shield at side is steel jacket with borated water thermally insulated from reactor tank. There follows an air-cooled space and then concrete-lined face of granite reactor pit. At bottom there is also a series of 4 steel plates, the uppermost being the one that supports the fuel lattice. Heavy water can be drained to two storage tanks at a lower level.

Heat. Reactor produces 10–20 tons of 230° C D_2O steam per hour. Steam emerges through two 4.4-in.-i.d. outlets in tank lid and goes to 2-unit heat exchanger to raise H_2O steam that may be offered on a "when available" basis to nearby plants.

Control. Control and shim rods contain enriched U and are operated hydraulically from below. Emergency control rods (sealed-off Cd-lined steel tubes) are supported from below and have positive buoyancy. Estimated 9% excess reactivity of reactor when cold makes necessary about 25 stations for control, shim, and emergency.

DEVELOPMENTAL FAST-NEUTRON BREEDER. Michigan (Atomic Power Development Associates project), 300 Mw heat, 90–100 Mw electricity, enriched fuel, Na cooled, 1959 (491, 816).

Description. Fuel is in $30\frac{1}{2}$-in-high part of vertical plates in boxes in $8\frac{1}{2}$-ft-high assemblies that have upper and lower blanket sections each 18 in. high. Blanket sections consist of depleted U in rods. Boxes are 2.664 in. on a side. Radial blanket rods are in $5\frac{1}{2}$-ft-high rods in hexagonal boxes in assemblies $8\frac{1}{3}$ ft high; these boxes are 3.398 in. across flats. Both core and blanket boxes have handling heads and lower cylindrical extensions that fit into support grid. Core has diameter of ~3 ft; radial blanket is 2 ft thick. Coolant level is maintained 11 ft above core to dissipate decay heat during unloading and to provide convection cooling in case of pump failure. Metal-sheet thermal shield is located on inner wall of 10-ft-diam. × 14-ft-high reactor vessel to reduce radiation damage to vessel. Further radial shielding, outside 1-in-thick vessel, consists of gas-cooled borated graphite followed by Borax ($Na_2B_4O_7 \cdot 10H_2O$). Following space for heat exchangers and pumps there is further biological shielding affording protection against 0.3-c/cm² induced activity in Na coolant. An airtight 84-ft-diam. cylindrical building having a $1\frac{1}{2}$-in. wall thickness contains the reactor.

Heat. Core and blanket are cooled by 13.2×10^6 lb/hr of Na (11.2×10^6 lb/hr through core) entering bottom inlet at 550° F and leaving top at 800° F. Na flows to heat exchanger where it heats 16.1×10^6 lb/hr of NaK from 500° F to 750° F. This goes to steam generator to heat 10^6 lb/hr of 400° F water to make 730° F steam at 600 psig. No isolating valves are used in the primary or secondary systems since they are considered more vulnerable than components they would protect. Syphon break on piping at reactor prevents drainage of the core if piping ruptures. If reactor vessel breaks, primary shield tank contains the Na. The 15,000-gal Na pool above core contributes largest part of 40-sec cycle time for Na flow in primary system, but minimizes thermal shocks to piping during scrams; also, pipe walls are only $\frac{3}{8}$ in. thick to lessen thermal shock.

Control. Boron shim rod is in core. Very little reactivity need be accommodated by this rod so neutron loss is small. Boron safety rods contain some 6% Δk; they are scrammed by gravity with an initial spring assist. During normal operation they are entirely above the core.

Research Reactors

Although details of many previously described (NU, March '52, p. 10; June '53, p. 65) reactors were given at Geneva, much interest, particularly among nations with reactors, centered on the four research reactors unveiled by Russia. Their features are summarized on page 53. While there is nothing startlingly different about these reactors, they do indicate the style of Russian reactor design and USSR research potential.

The first one, designed in 1947, was used for much of the USSR neutron research reported at Geneva. Reconstruction is planned to increase power level and shielding; ϕ will be raised to 2×10^{13}. RPT's core configuration is similar to that of APS-1. The third reactor, RFT, is a variant of pool-type reactors. Apparently, its principal use has been in shielding studies—there are extensive provisions for placing samples close to the core. RFT is based on a design considered and abandoned in early stages of RPT. The last reactor, which apparently is not completed, is very similar to RFT, although it has a higher flux and more beam holes.

Russia is designing 2-Mw research reactors using 10% U^{235} that it will sell to Poland, Czechoslovakia, Rumania, Hungary, Bulgaria and East Germany. Red China will buy a 6.5–10-Mw research reactor using D_2O and 2% U^{235} in rods.

Nations interested in acquiring research reactors found the detailed descriptions of specific reactors* of considerable interest, but a general orientation discussing advantages and disadvantages of the various types helped many to choose the best type research reactor for their needs.

Orientation on Research Reactors

L. Kowarski presented a survey to orient those who are contemplating acquiring a research reactor (*946*). Many different kinds of institutions are interested in research reactors. They have different research goals and reactor choice must be made accordingly.

Technological Factors

Kowarski outlined performance and design factors that should be considered by those choosing a reactor. Three factors, flux, power, and facilities, will be discussed here.

Neutron density. Flux is an important parameter for almost all uses of a research reactor. The most powerful reactors existing at present produce a maximum thermal neutron flux of a few times 10^{14} n/cm²/sec; zero-power reactors may produce as little as 10^7.

Power. High fluxes (useful for research) are connected with high power (useful for most other purposes) but the two are not coincident. In a *given* reactor, the flux observed at a given moment *is* directly proportional to the total power which the reactor produces at that moment. But two *different* reactors run at the same power may produce very different flux values.

The relevant factors are: (a) the total volume of the reacting medium (power produced per unit volume is what counts for flux); since power as such is of little interest for research purposes, compact designs are preferred; and (b) the lifetime of the neutrons within the medium. Power level determines the rate of their birth, but population also depends on the rate of disappearance. An irreducible minimum is set by the fission process and the number of fissionable nuclei. Neutrons are also removed by capture in nonfissionable materials and by leakage.

The search for high performance in the design of a research reactor is inseparable from a certain number of compromises. There is: (a) the conflict between highest power extraction rate (efficient cooling devices) and the shortening of neutron lifetime due to these devices; (b) a very compact reacting core loses many neutrons through leakage; (c) every attempt to use the flux reduces the maximum flux; we have a choice between a higher flux available for fewer uses, and the contrary.

Experimental facilities mostly take the form of channels leading through the shield and reflector towards the edge or inside of the core. Most of these channels have special equipment and are specialized according to their intended use. The total number of openings should be at least of the order of 10; in less compact types this number may go as high as several dozen.

Reactor Types

Kowarski summarized characteristics of the principle research-reactor types. They are listed below with their power, flux, and comments. Further details can be gotten from published descriptions and from NU, March '52, p. 10 and June '53, p. 65.

Uncooled, lowest-energy reactors. Using graphite and natural U: CP-1, Chicago; GLEEP, Harwell. With D_2O and natural U: ZEEP, Canada; ZOE, France. Unshielded (for powers of a few watts, fluxes of 10^7) or with some shielding to allow operation at a few kilowatts.

Such zero-power reactors can constitute either the easiest introduction to the reactor technology in general or a first step towards the construction of a novel type. In latter case, running at very low power may be required so that components do not become radioactive.

In the present state of information and supply, it is unlikely that more such reactors will be built for research. However, the use of first-step "critical assemblies" is likely to develop. Some past examples are: LOPO, Los Alamos, a prelude to the development of the water-boiler type; more recent critical experiments with Pu solutions; Dimple, a D_2O and U assembly at Harwell.

A zero-power reactor requires up to a factor of 2 less fuel and moderator than a full-scale reactor of a similar composition provided with cooling and experimental facilities. (More)

* Some of these are included in UN papers 5, 6, 401, 402, 404, 485, 486, 487, 488, 489, 762, 804, 859, 860, 915, 1072. Titles and authors are given on p. 94.

Graphite and natural U. X-10, Oak Ridge; BEPO, Harwell; Brookhaven; research reactors being built presently in Belgium and Western Germany. Power: 3,000–30,000 kw. Flux: around 10^{13} (highest value, reached at Brookhaven, is $\sim 5 \times 10^{13}$). Experimental facilities: very numerous and convenient, owing to great bulk and ease of access to core.

As enriched U and D_2O become available, it is unlikely that many more such reactors will be built for research purposes alone, since they are far too costly in ingredients and cooling machinery. Graphite and natural U remain, however, of interest for power production, and pilot plants of this type may be provided with research facilities. It is also to be noted that most radioisotope production, for which high fluxes are not necessary but great volumes available for irradiation are useful, is done in reactors of this type at present.

D_2O and natural U, cooled by convection. CP-3, Argonne; JEEP, Kjeller, Norway; Chatillon (replacing the French uncooled reactor); SLEEP, Stockholm. Power: $\sim$100 kw. Flux: ≤ 1–2×10^{13}. At this flux level radiation-induced decomposition of D_2O is significant enough to necessitate circulation of tank gases though a catalytic recombiner.

A factor 3 or so in power and flux could probably be gained by realizing all the potentialities of this cooling system; such a D_2O and natural U reactor, still simple, might compete with H_2O and enriched U systems in the 10^{13}–10^{14} flux region. If such a competition does take place, it will be governed by the availability and cost of D_2O and enriched U.

D_2O and natural U, cooled by compressed gas. Only example: P-2 Saclay, France. Max. power: $\sim$2,000 kw, $\phi_{max.} = 8 \times 10^{13}$.

Gas cooling, as shown by the British plans for energy production, can probably be made 3 or 5 times more efficient, reaching fluxes of a few times 10^{14}. For research purposes, however, cooling by water is probably more convenient at these flux levels.

D_2O and natural U cooled by H_2O. Only specimen: NRX, Chalk River. Power: 40 Mw, $\phi_{max.} = 7 \times 10^{13}$, access to numerous regions in which the flux is well above 10^{13}.

For many years, this very unique reactor offered world's highest flux and still is remarkable for its numerous, large, and convenient experimental facilities. However, it is more costly than most modern reactors and its power is too high according to safety standards valid for most places in which a research reactor is likely to be located. Great bulk, although not likely to be reproduced in modern designs, is advantageous to experimenter.

D_2O and natural U, cooled by forced circulation. NRU, Chalk River, 1956, a further development of NRX type. Power: 200 Mw. Flux: 3×10^{14}.

A Pu-producing power prototype, NRU cannot strictly be considered in research-reactor class. But, it is noteworthy that such multipurpose reactors offer fluxes not much smaller than those reached in the most efficient existing research reactors and that elaborate research facilities, mostly for reactor technology, can be provided. This provision for research in nonresearch reactors may yet become an important future trend.

D_2O and slightly enriched U. Recently completed design (EL3, France) uses ½ ton of fuel with twice natural amount of U^{235}. Power: 15 Mw. Max. flux: 10^{14} in center of 3-m^3 active volume. D_2O under forced circulation is both moderator and coolant. Design was dictated mainly by fuel supply.

Water boiler. SUPO, Los Alamos; North Carolina State College. Commercially available. Power: SUPO is 50 kw. Flux: at 100 kw, about 3×10^{13}, favorable fast to slow ratio. Experimental facilities: thermal columns, beam ports, and experimental channels. In view of small core volume, the boiler is best suited for beam experiments.

Cost is $100,000–500,000; lowest figure is for a minimum of experimental facilities; in more elaborate versions, facilities constitute the major cost item.

Swimming-pool. Bulk Shielding Facility, Oak Ridge, and numerous reactors in construction, 1956. Commercially available. Flux: at 100 kw, ϕ inside the core $\leq 3 \times 10^{13}$ (nearly unavailable for experimentation); at surface of core (available) 10^{13}. Fast flux at core surface is about ¼ of slow flux. Facilities: Oak Ridge Swimming Pool was designed mainly to offer tests with bulk shields, which could be simply immersed in water. Water-proofed tubes leading to core vicinity also can be provided. In later versions, core is movable between central position (for bulk experiments), and a corner position in which core is close to wall. The usual beam holes can then be provided reaching through concrete and wall to the immediate vicinity of the core. In a design proposed by Bendix, the proximity to wall is generalized to the whole perimeter; this represents a transition from the "pool" to the "highest flux" type (see below).

Published figures indicate the simplest version of the swimming pool may cost less than a well-equipped water boiler. Pool-type reactor is the simpler tool, suited for bulkier and less refined experiments than those performable with the water boiler. But complicated variants could make the swimming pool cost definitely more than the water boiler and will offer appreciably more varied research possibilities.

D_2O and highly enriched fuel. CP-5, Argonne. $\phi_{max.} > 10^{13}$ at 1,000 kw; falls slowly to 0.5×10^{13} at tank edge and affords vast experimental volume with about 10^{13}. Available fast (virgin) flux: about 10^{12}.

CP-5 provides a high (but not the highest) flux, with large number of experimental facilities and relatively low power. The future of this type depends on the possibility of running at higher powers. British E-443 is intended to run at 10 Mw and reach 10^{14}, but requires a 3-kg fuel charge.

Highest fluxes. MTR, Arco; reduced versions: LITR, Oak Ridge, and ORR, Oak Ridge, 1956. Flux: MTR at 30 Mw, max. (in reflector) 4×10^{14}; avg. available, 2×10^{14}; avg. fast (virgin), 1×10^{14}. ORR may reach 10 Mw, with 10^{14} available flux. Facilities: MTR has usual types some of which go through highest-flux region; ORR projects provide channels through core. MTR costs 5–6 times more than ORR; this may point to a "law of diminishing returns" for present design at highest powers and fluxes.

Although D_2O-moderated reactor affords best possibility of producing a very high *slow* flux, designers aiming at highest possible fluxes should not forget the claims of fast-neutron research; here H_2O asserts itself again. As regards cooling at highest powers per unit volume, both waters are theoretically equivalent; in practice, D_2O demands additional guarantees against pollution and loss by leakage, so that the already harassed designer may prefer to deal with H_2O.

Russian Research Reactors

HEAVY-WATER RESEARCH REACTOR. Russia, 0.5 Mw, natural U, D_2O moderated and cooled, $\phi = 2 \times 10^{12}$. Built for physical research and heavy-water experience. Critical in April, 1949 (623).

Description. Fuel consists of ~63-in.-long U rods suspended from a rotatable plate. This permits aligning fuel elements with holes in the top shield for loading and unloading. 92 gpm of D_2O coolant at 20–43° C enters at bottom of core tank, rises past fuel rods by natural convection, and is drained from the top at 38–65° C. One of two 14.7-cfm pumps circulates the D_2O through stainless steel heat exchanger cooled by 177–270 gpm H_2O. Helium at 1.1–1.45 atm is maintained above the D_2O and is circulated at 46 cfm by blowers to remove 0.05% (in He) radiolytic gases. The D_2O is contained in a 70-in.-diameter 78-in.-high ½-in.-thick Al-sheet tank surrounded at side and bottom by a 3.3-ft graphite reflector. Then there is a sealed tank and an 8.3-ft concrete shield.

Control. There are 4 gear-rack-driven Cd control rods. Their positions can be monitored to within 0.004 in., corresponding to the introduction of 0.1 mg of B into the reactor. There are 2 electromagnetically latched safety rods and an emergency D_2O dump device. Temperature coefficient of reactivity = $-1.4 \times 10^{-4}\ \Delta k/°$ C.

Facilities. There are 19 channels for irradiation and 10^7-n/cm²/sec neutron beams; 3 of these extend through the graphite reflector to the Al D_2O tank. There is a 4.5-ft-square graphite thermal column extending to the reflector. Because of the considerable excess reactivity available if all fuel elements are used, it has been possible to eliminate fuel rods from a 2-ft-diam. cylindrical region about the core's axis. The 2×10^{12} flux here is about as high as anywhere in the reactor and is well thermalized. A 3.5-in.-diam. 0.080-in.-wall Al tube extends down to this region from the roof of the reactor to permit irradiations or beam extraction. Other irradiation channels are at top and sides. Much basic physical research has been done with this reactor (*658*, *650*, *660*).

REACTOR FOR PHYSICAL AND TECHNICAL INVESTIGATIONS (RPT). Russia, 10 Mw, enriched U, graphite and water moderated, water cooled, $\phi_{max} = 8 \times 10^{13}$ n/cm²/sec. First operated April, 1952 (620).

Description. Fuel elements are hollow cylinders of enriched U in Al casings; 37 of them fit into 2.1-in.-i.d. Al-lined ducts on a square lattice with 5.5-in. spacing in the 1.8 gm/cm³ graphite moderator. Core is ~39 in. high and ~39 in. diam. The side and top graphite reflectors are 2.6 ft thick; the bottom reflector is 2 ft thick. Cooling is by 150 ft³/min of ~25° C distilled water that flows from the top down along the space between element and the duct in which it is placed and up through the center of the hollow fuel elements and out at ~60° C. Maximum heat flux is 2×10^6 cal/m²/hr. Distilled water exchanges heat with river water in a stainless-steel system. Graphite temperature is <470° C. To transfer heat from graphite to reactor vessel, He slightly above atmospheric pressure is used. The cooling water has 46% the slowing power of the graphite core. Shielding at the sides is provided by a 1-in. Fe thermal shield followed by 10.5 ft of 2.4-gm/cm³ concrete. Top shielding is a 5-ft graphite layer followed by a 1.3-ft lead layer followed by an 0.6-ft Fe slab.

Control. All control rods operate from top of reactor. Two automatic regulating rods in reflector vary k_{eff} by 0.1%. Three manual and 3 slow automatic boron-carbide shim rods in core provide combined control of 10% k_{eff}. Two Cd safety rods in the reflector reach their extreme position in 0.4 sec and produce 1% change in k_{eff}. Three pneumatically operated Cd rods capable of reaching their extreme position in 0.02 sec were provided in anticipation of sharp reactivity changes caused by air bubbles in the cooling water; however they proved unnecessary in practice and were removed.

Facilities. There are 5 large vertical channels in the reflector for loops for testing components for new reactors. Fuel elements for APS-1 (p. 40) were tested here. These channels are presently devoted to test setups cooled by gas, liquid metal, high-pressure water (2 of these), and medium-pressure water. There are 6 smaller channels for testing fuel elements and for irradiations. For research, there is a 1-meter-square 6.5-ft-long graphite thermal column with 10^7 n/cm²/sec. Two 4-in.-diam. 10^8-n/cm²/sec beams emerge from the core; one of these is used for a long-path mechanical velocity selector. There is a 10^8-n/cm²/sec 2.4-in.-diam. beam from the reflector.

EXPERIMENTAL NUCLEAR REACTOR WITH ORDINARY WATER AND ENRICHED U (RFT). Russia, 0.3 Mw, 10%-enriched U, H_2O moderated and cooled, ~2×10^{12} n/cm²/sec mean flux. Used for shielding studies, isotope production, and physical research (621).

Description. The fuel loading consists of 32 units fixed at top and bottom by Al grids; 24 units have 16 fuel rods, the remaining 8 have 15 so as to provide space for control rods. The Al-canned 10%-enriched fuel rods, 0.35 in. o.d. and 20 in. long, are arranged in a square lattice with ~0.7-in. spacing. The core is quasi-cylindrical, 16 in. in diameter and 20 in. long, with a total loading of ~3.5 kg of U^{235}. Entire core is located off center at bottom of an Al tank. Core is cooled by ~150 cfm of 30° C water in closed primary circuit that heats up about 1° C after passing the 70° C fuel elements; 10% of the cooling water is passed through a deaerator to remove radiolytic gases. Activity in exit water from each unit is monitored to detect broken fuel elements—none have been experienced. Secondary circuit has 30-cfm flow. Shielding is mostly water and cast iron. On the north wall there is a large plug that can be rolled back on a truck to permit access to high-flux regions within. Two cast-iron plates can be lowered into place to provide shielding while the plug is withdrawn.

Control. Control is by means of a steel rod, servo controlled. For safety there are 3 boron-carbide rods with solenoid release that complete their free fall in ~0.5 sec. Scram is initiated if preset power level is exceeded by 20%, start-up velocity is intolerably high, primary water flow drops off, or power to circulation pumps fails.

THERMAL RESEARCH REACTOR (TRR). Russia, 2 Mw, 10%-enriched U, water moderated and cooled, $\phi_{max} = 2 \times 10^{13}$, not built yet (622).

Description. Core is composed of 52 vertical units containing 4.5 kg U^{235}, most units contain 16 fuel rods, but 9 contain one less to provide space for control rods. Fuel portions of the rods are 20-in.-long and are in 0.4-in.-o.d. Al tubes. They are located on a square lattice with 0.7-in. spacing, fixed by Al grids at top and bottom. Core assemblies are in a 2.16-ft-diam. Al tank located eccentrically in a higher water-filled 3.5-ft.-diam. tank. 4,000 gpm of water flow up through outer tank and down through core tank removing up to 3.8×10^8 cal/m²/hr from fuel surface. Fuel element surface temperature is 90° C; water temperature in core averages 35° C. Secondary 1,100-gpm water circuit removes heat from primary system. Side shield is water, cast iron, and concrete.

Control. There are 9 control rods, consisting of 1 steel regulating rod, 5 boron carbide shim rods, and 3 boron carbide safety rods.

Facilities. There are 11 radial horizontal holes extending to the core edge; these have $0.5–1 \times 10^9$ n/cm²/sec at their outer ends. A horizontal graphite thermal column extends from side of the eccentrically located core tank closest to the shield. A bismuth shield is between core and thermal column to attenuate gammas. There is a vertical biological hole in the shield on the side opposite the thermal column. Radiosotopes are produced in two vertical holes that extend down to the upper and inner edge of the thermal column.

REPORT from GENEVA

reactor fuels

Th and U purification—fuel systems—radiation damage—fabricating fuel elements—survey of reprocessing methods—solvent extraction

Reactor-Fuel Metallurgy

Preparation of U and Th metal has been correlated with reactor technology. A standard USA process is bomb reduction of UF_4 (*825*,* *817*). Steps are: hydrogen reduction of UO_3 (from ore refining) to UO_2, fluorination with HF to UF_4, reduction to metal with Ca or Mg in a CaO-MgO-lined steel pressure vessel. Mg is the preferred reductant; 5–7% excess insures yields of 95–98% U. Reaction bombs are heated at 550–700° C for 1½–3½ hr until the reaction, taking 1 min, occurs. Vacuum remelting and casting in graphite crucibles produces metal with 200 ppm C, and some Fe, Cr, and Ni.

British method. Crushed U ore is dissolved in mixed HNO_3 and H_2SO_4, the solution purified, and H_2O_2 added to precipitate uranium peroxide, which is filtered (*407*). Warm HNO_3 dissolves the peroxide, and single-stage ether extraction removes remaining impurities. Then NH_3 is used to precipitate ammonium diuranate. This is heated to decompose it to UO_3, reduced to UO_2 by H_2, and treated with HF to form UF_4. Further operations are similar to USA practice, but Ca is reductant. Exothermic reaction is triggered by dropping burning pellet of KNO_3-lactose into bomb.

TABLE 1—Metallic U Fuel Elements

Type	Dimensions: length × diameter (in.)	Jacket* thickness (mils)	Bond	Temperature (° C): Surface	Temperature (° C): Center	Maximum: Heat flux (watt/cm²)	Maximum: Specific power (watt/gm U^{235})
X-10†	4.06 × 1.1	Al-20	No bond	245	349	2.3	24.8
X-10	4.07 × 1.36	Al-35	Al-Si alloy	245‡	349‡	2.3	24.8
CP-3	66 × 0.85	Al-59	No bond	43	49	4.2	168
EBR	7.5¶ × 0.364	SS-22	Liquid NaK	335	391	240	38.8
SRE	266¶ × 0.75	SS-10	Liquid NaK	538	649	107	415

* Final closure of jacket is by shielded-arc weld, except for seam-weld, crimped end used on first type. Shielded-arc closure is also used on 10-mil mild steel cans that jacket Pu-metal fuel slugs for Clementine. Slug is 5.5-in. long × 0.65-in. diameter (*825*).

† Variation has 35-mil jacket and is shielded-arc closed; length is 4.07 in.

‡ Bonding should reduce temperature, but thicker jacket and more U work against this.

¶ In EBR single U slugs, 1.9-in. long, are positioned inside SS tube; in SRE single slugs are 6-in. long.

Purifying Thorium

Russian procedure for Th starts (*635*, *636*) with gravity and magnetic separation of monozite from sand. Sulfatization with H_2SO_4 or treatment with NaOH or $NaCO_3$ results in a Th and rare earth mixture. Ways to separate this are: thorium hydroxide precipitation by partial neutralization and preferential dissolution (10^4 increase in Th concentration, to 90%); preferential crystallization of the sulfate—for direct ThO_2 production, or of the oxalate—for further electrolytic refining; use of special sorbents to form very high purity Th; and extract from HNO_3 solutions with tributyl phosphate.

Metallic Th powder is made by electrolysis of fused NaCl and KCl salts with a liquid or solid cathode, or by Ca reduction. ThF_4 electrolysis with a solid cathode is most used. Optimum conditions are 800° C (60–70 amp/kg of electrolyte), cathode current density of 20 amp/dcm², and anode current density of 0.5 amp/dcm². Current efficiency and metal yield are 70–80%.

Alternate to electrolysis is reduction of ThO_2 with 25% excess Ca, and 40% (of thoria) $CaCl_2$ as fluxing agent. Temperature is 1,000–1,100° C. Water and dilute HCl dissolve the slag, and the Th powder is treated with 15% HNO_3 (7% for electrolytic Th), then

* References to UN papers on p. 94.

2% HNO_3, followed by water wash, and drying with alcohol and ether. The content of final powder is 99.5–99.8%. Final fabrication is by pressing (powder has good pressibility and high plasticity) or sintering at 1,100° C, followed by cold coining and annealing at 1,100° C under vacuum. See (*531*) for USA paper on theory of U and Th reduction.

Fuel Systems

Goal of fuel-system designers is to maximize life and specific power of fuel elements (*825*). Minimum stable life for fuel should be 2% burnup, or about the life as limited by poisoning in thermal reactors. Specific power may be limited by heat flux to coolant (surface temperature or nonboiling limit) or by heat flux and temperature within fuel (central temperature, or thermal stress

TABLE 2—Fluid Fuel Systems

Fluid	*Fuel**	*Concentration (gm U^{235}/l)*	*Temp. (° C)*
H_2O	UO_2SO_4	30–40	242–250
H_2O	$UO_2(NO_3)_2$	45–50	20–78
H_2O	$(UO_2)_3(PO_4)_2$†	100–150	20–455
D_2O	UO_2SO_4	4–5	242–250
Bi	U	0.05–0.66‡	355–510
Bi	UBi_2 (slurry)	<10‡ max.	510 max.
Na	UO_2	<30‡ max.	870 max.
H_2O, D_2O	U oxides	Problems: Stability of suspension, erosion of system	
NaK	UO_2	Erosion of system, settling	
He	UO_2	Possibly erosion and handling	
Bi	U_3Bi_2	Erosion, maintain particle size	
Fused fluorides	UF_4	Corrosion	

* Container material is stainless steel for first four systems; low Cr, Mo steel for systems 5 and 6; and austenitic steel for system 7 (*825*).

† This is a Los Alamos power reactor experiment. System is interesting because there is no discontinuity in solubility across the critical point of water. Only Au and Pt withstand corrosion. Au-plated SS is used.

‡ These figures are wt % of U. BNL is investigating the system (NU, July '54, pp. 11–42). Mg is used as O getter, and Zr to inhibit mass transfer.

TABLE 3—Dispersions of U in Matrices and Ceramics

UAl_4 in Al—Plates 60-mil total thickness including 20-mil Al coatings, 3-in. wide, 2-ft long, are used in the MTR and the ORNL swimming pool. UAl_4 is 10–20 wt % of core (*955*). MTR uses 414 curved plates, 19 plates/element, with 0.115-in. water flow passage between each. Surface temperature is 116° C, heat flux is 195 watts/cm², specific power is 9,400 watt/gm U^{235}. Surface temperature for swimming pool is 93° C, flux is 32 watt/cm², specific power is 2,000 watt. CP-5 uses 10 plates, dimensions not given, with 0.154-in. water spacing. Swimming pool uses 18 plates with 0.117-in. spacing (*825*). For plate element fabrication see p. 58.

UO_2 in Al—Geneva conference reactor uses 18 plates containing 54% UO_2-remainder Al powder, with 0.117-in. water gap between plates.

UO_2 in stainless steel—Flat plates 30-mil thick, with 5-mil coatings, 2.5-in. wide × 23-in. long are used in NAA pool-type reactors. Calculated conditions are 297° C center temperature, 70 watt/cm² flux, and 2,000 watt specific power.

U in Th—Cylinders of this material, containing 4% U^{235} would have same dimensions as SRE fuel in Table 1. Center temperature is 871° C, flux is 274 watt/cm², specific power is 694 watt.

UO_2 in BeO—Proposed ceramic dispersion for high-temperature reactor is BeO tube, impregnated with UO_2. Suspend ¼-in. thick × 1-in. i. d. × 4.5-in. long tubes inside BeO-moderator bricks.

UO_2 in graphite—For low-power reactor use, hexagonal graphite block impregnated with U_3O_8; 8.37-in. diameter × 4-in. high.

UC_2 in SiC—For high-temperature gas turbine plant use, ceramic blocks of SiC plus UC_2.

UO_2 in ThO_2—For use in thermal breeder reactors.

UO_2 in austenitic steel— For small water-cooled and moderated plants. In high-temperature systems, thermal fatigue limits use.

TABLE 4—Characteristics of Aqueous ThO_2 and $UO_2{\cdot}H_2O$ Slurries (*811*)

Crystal form

—ThO_2. Crystals (200 Å) exist as agglomerates (1–10 microns)

—$UO_2{\cdot}H_2O$. Rodlets: 1–5 micron diameter, 10–30 microns long. Platelets: 6–50 microns on edge, 1 micron thick

Preparation

—ThO_2. Calcination of oxalate or formate at 650–800° C

—$UO_2{\cdot}H_2O$. Rodlets: Heat anhydrous trioxide, formed by heating uranium peroxide at 300° C, in water at 185–300° C. Platelets: hydrate trioxide, made by decomposition of uranyl nitrate at 300–400° C, at 185–300° C

Sedimentation

—ThO_2. Satisfactory up to 300° C; easily redispersed; no severe packing

—$UO_2{\cdot}H_2O$. Satisfactory; platelets settle faster with slightly higher bulk density

Caking and viscosity

—ThO_2. Caking on pipe walls occurred during circulation at 250° C when impurities present; below 200° C plastic mixture obtained with many slurries at room temp. after violent agitation. Lower the calcination temp., more readily slurry becomes plastic with violent agitation—yield stress for slurry oxide calcined at 800° C is ½–¼ that calcined at 650° C. Oxalic acid or other additives reduce drastically viscosity of plastic slurries at room temp. (10 millimoles oxalic acid per mole thoria reduces yield stress of 600-gm/kg slurry from 0.37 to zero lb/ft²)

—$UO_2{\cdot}H_2O$. Principal difficulty in circulation tests was caking on pipe walls at elevated temp. Rodlets apparently transform into platelets that cake out at 250° C. Cakes not formed up to 200° C. Tendency to cake reduced greatly by additives. No difference in viscosity between rodlets and platelets at 250 gm U/l (0.6 centerpoises at 71.8° C, 0.96 at 39.5° C)

Attack on materials

—ThO_2. Abrasion-corrosion troublesome in regions of high fluid acceleration; at 20–25 ft/sec, attack on stainless steel absent in straight pipes and standard elbows. Slurry calcined at high temp. attacked s.s. more rapidly than low-temp. calcined. Materials forming adherent oxide film (Ti, Zr, Zircaloy-2) resistant to attack. Stellite 98M2 and Carbaloy did not wear well—possibly due to attack on cobalt

—$UO_2{\cdot}H_2O$. Max. velocity 83 ft/sec at throat of orifice; in straight pipes and elbows, 20 ft/sec. Attack nil on stainless steel at 250° C

in fuel or in jacket). The tables summarize proven and tentative USA fuel systems (pp. 54 and 55). Fuel fabrication is shown on p. 58.

Slurry Fuels

Aqueous slurries, less corrosive than solutions and having as high or higher neutron economy, are promising for fuels or breeder blankets (*811*, ORNL).

Of possible thorium compounds, the oxide is the most promising. The more difficultly prepared fluoride or oxyfluoride slurries hydrolyze into gelatinous mixtures at high temperatures. Carbonate decomposes in water at high temperature. Nitrate with N^{15} merits attention, except for the thermal and radiation instability of nitrates in general and the expense of N^{15}.

Because of the caking tendency of $UO_3 \cdot H_2O$ at high temperature, other slurry compounds were tried. Carbonate is unstable at high temperature; excess CO_2 gas must be present to prevent decomposition. Magnesium diuranate forms soft cakes on pipe walls at 250° C; slurry *p*H of above 10 would enhance attack on stainless steel if oxygen were present in the slurry.

In a test circulation loop, the Westinghouse 100A, canned-rotor, 3,450-rpm, centrifugal pump was used at high temperatures. Designed to deliver 100 gal/min with a head of 272 ft, it has operated successfully with slurries at 300° C and 1,700 psi. The bearing-journal assemblies, Graphitar and stellite 98M2, offer little difficulty if they are continously flushed with water containing no solids.

Some results are given in Table 4 on page 55.

Radiation Damage in Reactor Fuels

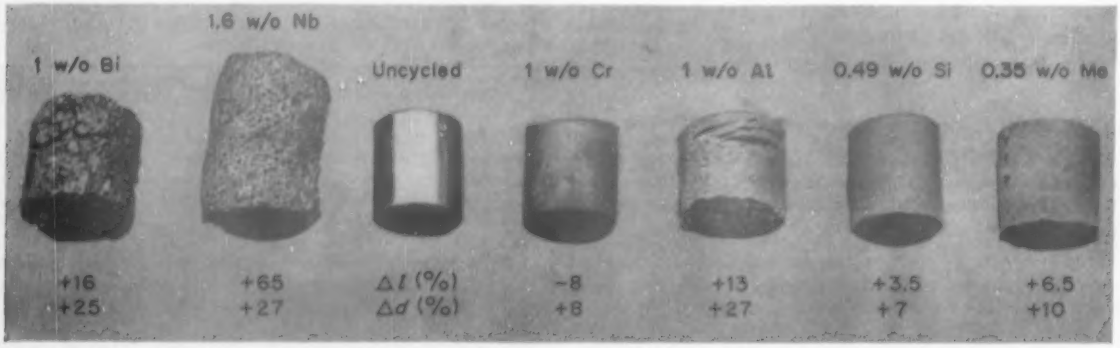


FIG. 1. After 500 thermal cycles from 200–700° C various U alloys showed these changes. Improved stability of alloys results from more sluggish, possible absent, α-β transition; higher strength of α phase; finer grain size; and more random orientation. Thermal-cycling is used as screening test, because thermal-cycle failures inevitably fail under radiation (829)

Radiation damage is caused by interactions of fission fragments with fuel atoms, and by elastic neutron collisions. In U there is a superficial resemblance between thermal-cycling effects (Figs. 1–6) and irradiation. Radiation distortion is related to size and orientation of the metal grains, and fabrication and heat treatment techniques (*744*). Changes in engineering properties of U and Th are given below. Burnup of 1% corresponds to 10,000 Mw-day/ton exposure of natural U.

Radiation damage theory. Theories on mechanism of radiation damage in fuels have developed along similar lines in the USA (*744, 745*), UK (*443*), and USSR (*681*). All agree that complete theory is not yet developed. Experimental data from the three seem to compliment rather than repeat. Plasticity of U during irradiation was reported (*681*). U creep increased by factor 1.5–2. In one experiment 75% enriched U was placed on a Mg core, then the core and U were pressed into a stainless steel tube. After 10^{19} nvt, the U was strongly wrinkled at the Mg-U boundary. U thickness varied from 0.07–0.45 mm, compared to 0.3 mm in smooth parts.

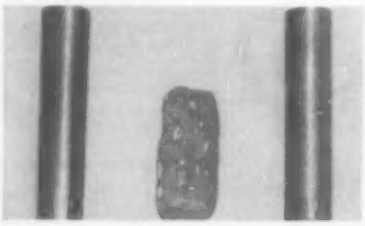


FIG. 2. Changes in α-rolled U caused by going through 660° C transition, which causes 1.5% volume increase. After 200 cycles from 580–630° C Δl is 0.7%, Δd −0.4%; 630–680° C Δl −31%, Δd 19%; 680–730° C Δl −2.6%, Δd 2.4%

U—U recrystallized at 575° C, and cold-worked to 70% area reduction shows ≈25% less growth at 0.11-mm grain diameter than at 0.04. Specimens rolled at 600° C, beta quenched, then worked at 300° C show 12% less growth than similar specimens without the beta quench, showing that rolling effects are additive. Specimens rolled at 600° C or higher don't grow or show shrinkage. Rolling at 300° C allows growth of 800 × 10^{-6} per ppm burnup. But recrystallization after rolling reduces growth (*745*).

U growth is an exponential function of burnup; some data suggest secondary dependence on irradiation temperature. Plot of *ln* growth rate vs $1/t$ shows 3.4–3.6 at 150° C and 2.8–3 at 60° C, in nonlinear variation. Surface distortion takes form of "orange peel" effect that destroys contours of large-grain random U. Fine-grain random U shows fine surface irregularity that roughens specimen. Density decreases only slightly, despite dimension changes; % decrease is 20 × % atom burnup (*745*).

At 0.035% burnup at 120° C, these are the changes in strength: −27% ulti-

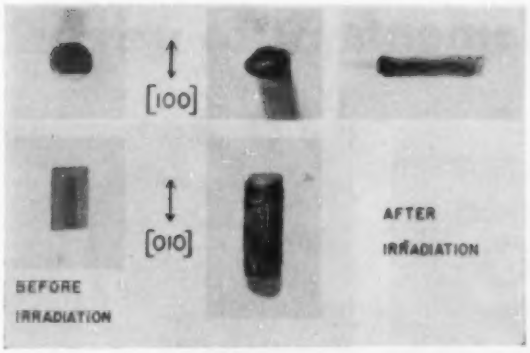


FIG. 3. Cylindrical single crystal of U before and after irradiation to 0.1% burnup. U is in the alpha form. Cylinder diameter was 0.125 in. (*745*). Growth coefficients are (in 10^{-4} per ppm burnup) direction (100) − 420, direction (010) + 420, direction (001) 0, all ± 20

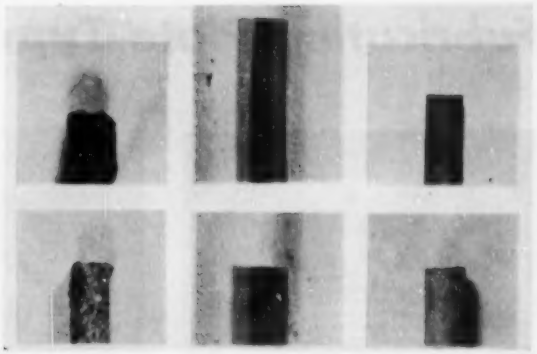


FIG. 4. U surface after 0.1% burnup for top—as cast, 300° C rolled, 600° C rolled; bottom—600° C rolled and β-annealed, 300° C rolled and β-quenched, 600° C rolled and β-quenched. Distortion of large-grain U is by "orange peel," while fine-grain metal has only fine surface irregularity

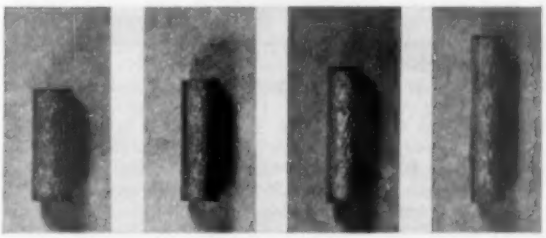


FIG. 5. Effect of 0.1% burnup on 600° C rolled, β-quenched U, after (l. to r.) no cold-work, 5% reduction, 10% reduction, and 25% reduction, all at 300° C. Work produced an (010) texture, which means increased radiation growth. Growth is less if work is at higher temperature

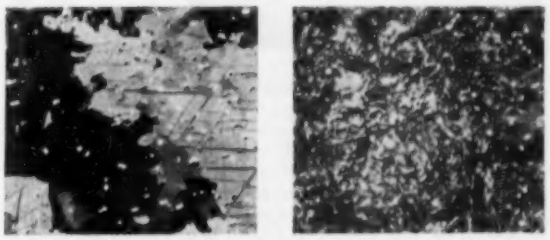


FIG. 6. Metallographic structure of β-quenched U, viewed in polarized light. Left—before irradiation, right—after irradiation. Grain outlines are badly distorted, as if by cold-work. "Cold work" is result of internal adjustment by interference of grains that have been restrained from growth

mate strength, +117% yield strength, −98% elongation, +10% Young's modulus (*745*). Diffusion between U and cladding materials is not aggravated by irradiation. Corrosion resistance is radically decreased for U_3Si, 5% Zr, 1.5% Nb, and 9% Mo alloys (*745*).

U single crystals—Irradiating single crystals of U at 200° C to 0.03% burnup caused 15% growth in the [010] direction; 500° C irradiation caused no growth. Cold-rolled U sheet grew the same amount. Indication is that 100% growth would be caused by 0.1% burnup (*443*).

Zero elongation strength resulted from 10^{19} nvt exposure of U regardless of heat treatment. Ultimate tensile strength and impact strength decreased by >2. Microstructure of irradiated U shows many twins. These are wavy. A particular globular structure of a U grain is seen at 15,000× magnification (electron microscope). Rolled U foil, both annealed at 600° C and unannealed, show 8% growth after 2.5×10^{19} nvt (*681*).

U-Al—Alloys with 5–17% U changed 1% in dimensions with exposures to 0.6% burnup. Thermal conductivity decreases linearly to ⅔ initial value. Stress strength rises to 7×10^4 psi at 2×10^{20} nvt. At 0.05% burnup, density decrease is 3% (*744*).

U-Cr—Excellent stability results from 0.1–0.2% Cr in U. Mo, Nb, or Zr also are effective, but in higher amounts only. U_3Si is also stable (*730*).

U-Zr—Alloys of Zr-U show good dimensional stability, with thermal conductivity loss of 5% at 0.1% burnup. Density decrease is ~1% (*744*).

U-Mo—Length increase of 0.61% Mo in U averaged 13% after 5×10^{19} nvt exposure, if preannealing below 500° C was done. If preannealing was at 500, elongation was 2%. Half the exposure caused elongation of 8%, to unannealed alloy, and 1% to 600° C alloy. High Mo alloy, 9%, had no growth at 2.5×10^{19} nvt (*681*).

U-graphite—Serious damage to properties of U-graphite systems occurs because of anisotropic nature of graphite (*744*).

U-Be—Be alloys containing 0.5–3% U show trivial changes in dimensions and density at 0.06% burnup. Thermal conductivity did not change for 20% U alloy during 100° C reactor exposure (*744*).

UO_2 and U_3O_8—X-ray studies show pronounced line broadening for 0.0003% burnup. At 0.03%, lines in U_3O_8 are completely removed. UO_2 looks promising (*744*).

UO_2-BeO—Samples with 2 and 10% UO_2 increased linear dimensions by 1%, dropped 30% in compressive strength and elastic modulus, after 0.0005% burnup (*744*).

UO_2-stainless steel—UO_2 particle size has large effect on hardness increase (*744*).

U-Th—Th alloys with 1–5% U show only slight changes in dimensions and density (*744*).

Th—Excellent radiation stability of Th is result of its superior isotropic nature. No growth is reported after 10^{19} nvt. Yield strength increases by factor 2 (nvt not given) (*744*).

Fabricating Plate-Type Elements

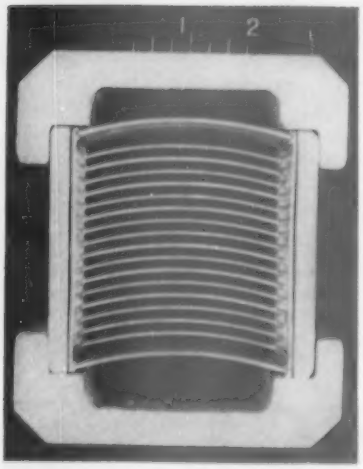


FIG. 7. Brazing is used to join plates into single uel assembly. Braze used is 12% Si-Al. Grooved side plates are braze-clad wrought Al sheet. Assembly is dried in metallic jig shown at 150° C for 3–4 hr, to drive off water and alcohol from flux; transferred to a ceramic jig and preheated at 500° C for 1 hr, brazed at 610° C in a forced-circulation furnace for 25 min; cooled to room temperature, washed; then cleaned in acid bath, hot-water rinsed, and dried

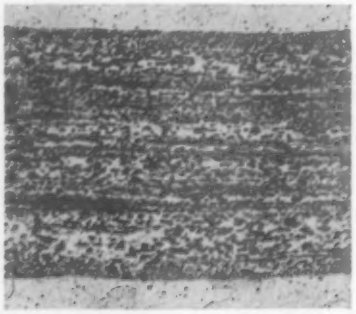


FIG. 8. Fuel plates contain central core of UAl_4 alloy, interspersed in eutectic of UAl_4 plus Al, surrounded by Al cladding. Technique used: Up to 25 wt % U is alloyed with 99.7% Al in a graphite crucible by induction heating in air. After casting and cooling, alloy is heated at 590° C for 1 hr, and hot-worked to ¼-in. strip stock, which is punched into blanks 2 × 2¼ × ¼. Up to 60 wt % U can be made by mixing −325 mesh Al powder and 44–105 μ UO_2, and pressing same size blanks under 33 ton/in². Blank is jacketed by frame and cover plates, all ¼ in. thick. Assembly is hot-rolled at 590° C in 6 passes to 0.07 in. thick. Next, surface is flux-annealed at 610° C for 1 hr. After flux is removed, plates are cold-reduced to 0.06 ± 0.001 in., and annealed for 1 hr at 590° C

FIG. 9. Incomplete MTR fuel assembly is made of 17 fuel plates, each 0.05 in. thick, plus 2 end plates 0.065 in. thick. Water flow gap between plates is 0.115 in. Water velocity is 30 ft/sec. End fittings of cast 356 Al alloy are plug-welded to adapt assembly to supporting grid in reactor, and provide flow passage. Other reactors with plate elements are Geneva reactor, ORNL swimming pool, and CP-5. For performance data on plate-type fuel element see Table 2, p. 55 (*953, 561, 562*)

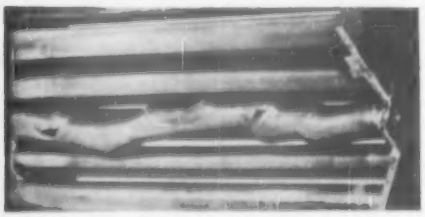


FIG. 10. Failure of natural-U fuel rod after corrosion testing in hot water was caused by intentionally induced cladding defect. PWR design calls for highly enriched U-Zr alloy plates clad with Zircaloy-2 for core, and UO_2 loaded into Zircaloy-2 tubing for blanket (*815*). Radiation-damage resistance of U-Zr is good (p. 57). For PWR details see p. 46

Heat Transfer Between U and Al*

Coefficient (watt/cm₂/° C)	*Treatment*
0.6	None
Doubled	Apply graphite
Unaltered	Mg and Al powder
2.0	Apply 25 atmos. pressure
3.5	Same for 2 wk
2.4–3.3	H_2 between gap
2.0–3.0	He between gap
1.9–3.2	Air between gap
25	Na, Pb, Bi, Mg liquid

* From (*947*) which concludes that pressure or liquid Na lining is best possibility. No tests were made on effect of Al-Si alloy bond (see Table 1, p. 54).

BNL Fuel Elements

BNL fuel elements are made of 1.1-in. diameter γ-extruded U, 4 in. long, inside a 11.5 ft. long, interior anodized, finned, 2S Al-tube. Anodizing is 1 mil thick. A butt cap is welded on one end of the empty tube, and the tube anodizing dehydrated. He leak testing follows. 33 U slugs are then loaded into the tube. End cap is then attached by brazing. A second leak test is followed by collapsing the tube around the slugs at 2,500 psi to get better heat flow. Tubing to carry He is then brazed to the brazed cap. Monitoring He pressure inside the element during operation serves as warning system for Al tubing failure. Third leak test completes assembling operation. Most failures have been in the He tube, usually near the brazed joint. Redesign uses welded cap and flame, rather than induction brazing, to join He tube to cap (828).

Survey of Fuel-Separations Processes

Separations methods for nuclear fuels must have high yields, low cost, and minimum holdup. Because remote fuel-element fabrication is not presently possible, complete removal of fission products is demanded. Cheaper schemes would be possible without this demand. Following are outlines of investigated methods (*823*). See p. 61 for solvent-extraction methods.

Precipitation of U and Pu. Successful plant-scale use of $BiPO_4$ precipitation has recovered 95% of Pu from irradiated U. Decontamination factors of 10^7 were attained. U is not recovered in the process. Reducing Pu to +4 and complexing U with $SO_4^=$ results in only Pu precipitation when $BiPO_4$ is added. Three cycles of fission-product precipitation with Pu as +6, and Pu precipitation with Pu as +4, are followed by a LaF_3 fission-product precipitation, a LaF_3 Pu precipitation, and metathesis of this precipitate with KOH. This product is dissolved in HNO_3, and final purification done by H_2O_2 precipitation.

Recovery of U can be by uranyl-ammonium phosphate precipitation, which has been pilot-plant tested. The method is simpler and cheaper for fuel solutions from some kinds of aqueous homogeneous reactors. Precipitation processes are flexible and easy to scale up, but not as adaptable to continuous operations or as capable of high recovery as other processes.

Ion-exchange concentration. Pu yields of 97% have been found in small pilot-plant runs using ion-exchange resins. Decontamination of 30–300 was obtained. Ion exchange is best suited for continuous processing of aqueous homogeneous reactors. New ion-exchange resins more resistant to radiation and chemicals have renewed interest in such purification (*550, 551*). It is superior to evaporation or precipitation for concentrating intermediate or final streams from solvent extraction. Concentration factors of 100-fold have been obtained, concurrent with 10-fold reduction in Zr, and some Ru removal. Silica gel is successful in final U cleanups (*539*). Organic and inorganic resins are both of interest.

Fractional distillation of fluorides. Demonstration tests have been encouraging on recovery of U using fluoride distillation. Pu remains with the nonvolatile fission products, making highly enriched fuels the most promising field. Decontamination factors are 10^7. If work in progress is successful in finding a way to extract Pu as hexafluoride from irradiated fuel feeds, the method would be superior for handling natural U fuels.

Method is based on low volatility of most fission-product fluorides compared to UF_6. Feed solution is prepared by reacting liquid-phase reagents, such as ClF_3-HF, BrF_3, or BrF_5, with metal fuel. Alloyed fuel elements or refractory fuels can be dissolved in fused salt through which HF, followed by F_2 or BrF_5 is dispersed. The resulting fluoride mixture is fractionally distilled. Only the fluorides of As, Te, Ru, I, and Mo boil below the UF_6 boiling point of 56.4° C. Extensive applications of fractional distillation are forecast.

Pyrometallurgical processes. Short-cooled, highly irradiated fuels can be handled by high-temperature processes (*542*). Processing of breeder-reactor blankets, to concentrate Pu, can be done in three ways.

Vacuum distillation from molten U at 1,500–1,800° C has removed 99% of Pu. Removal took 140 min at 1,680°, with 0.4-cm^2 surface area/gm of U. Major problems are materials of construction, and design of Pu collectors.

Salt extraction from molten U uses halides like UF_4 or $MgCl_2$. Equal weights of UF_4 and irradiated U resulted in 90% of the Pu entering the salt phase. For purifying fuel of U in molten Bi used in BNL's liquid-metal-fueled reactor, fused salt extraction is under investigation (*550*). Most rare-earth fission products are transferred into the salt phase, thereby removing many high-cross-section poisons.

Some metals, such as Ag and Mg, are immiscible with molten U and exhibit Pu distribution ratios of about 2. When using Mg, extraction equipment must be well designed because of its high vapor pressure. If Ag is used, subsequent separation of Pu is difficult, but can be done by Ag distillation (*545*). Fission products are not separated from either U or Pu by this extraction. Zone melting for purification has been explored, but rates of $< \frac{1}{2}_{50}$ linear in./hr are needed to realize the theoretical separations factors.

Reactor-core separations. In recycle of fuel from reactor cores, removal of fission products is the primary objective. U and Pu need not be separated. Two different approaches are being studied.

Oxidative slagging (*544*) in the presence of limited O_2 results in formation of a skim of rare-earth and Ce oxides floating on the surface of molten fuel. As in other pyrometallurgical operations, rare gases and halogens bubble off. Cs, Sr, La and Ba diffuse into the crucible and through the slag layer. Oxide crucibles, or addition of oxides, supplies the O_2. Adding graphite improves Zr removal. (See p. 63 for photos.)

Electrorefining depends on transportation of heavy metals through a molten bath of alkali or alkaline earth halides. Noble-metal fission products collect as an anode sludge. Rare gases are vented at the anode. Other metal fission products build up in the bath, which must be replaced or separately processed to keep decontamination high. U goes to the cathode, and by using a metal that forms a low-melting alloy with U as the cathode, U can be continuously removed as a liquid. On small-scale runs material recovery at the cathode represented 98% of the metal leaving the anode. Decontamination factors of 100–300 were obtained for Ce, Cs, Sr, Zr, and Ru.

Another Possibility

A pyrometallurgical process similar to the deBoer process might be used for core processing. U is iodized and then reclaimed as metal by hot wire (or tube) decomposition of the volatile UI_4. Only Zr and Nb seem to transfer with the U, but better cold-trap arrangements would give better separations. Fate of Pu has not been determined yet. Another application of the deBoer method is the separation of U^{233} from Th.

Distribution Coefficients

$\frac{\text{concentration in organic}}{\text{concentration in aqueous}}$ of U, Pu, and Fission Products:

URANIUM

Aqueous phase	*Organic phase*	*Distribution coefficient**
Uranyl nitrate 0.2 molal (*413*)	Diethyl ether	0.003
	Dibutyl cellosolve	0.001
	Dibutyl carbitol	0.01
	Penta-ether	0.12
	Methyl isobutyl ketone	0.004
	Cyclohexanone	0.17
	Isoamyl alcohol	0.004
	Tributyl phosphate	5.5
Uranyl nitrate 1.0 molal (*413*)	Diethyl ether	0.14
	Dibutyl cellosolve	0.026
	Dibutyl carbitol	0.32
	Penta-ether	1.0
	Methyl isobutyl ketone	0.16
	Cyclohexanone	0.9
	Isoamyl alcohol	0.15
	Tributyl phosphate	1.8
U (VI) in 1*M* $Al(NO_3)_3$¶ at *p*H 0.0	Methyl isobutyl ketone (*824*)	2.1
*p*H 0.5		1.6
*p*H 1.0		1.3
*p*H 1.5		1.0
*p*H 2.0		0.78
*p*H 2.5		0.6
U in 2*N* HNO_3 10 gm/l	40% tributyl phosphate (*549*)	5.7
50 gm/l		2.6
100 gm/l		1.45
250 gm/l		0.62
400 gm/l		0.39
U 10 gm/l HNO_3 at 0.6*N*		3.3
1.5*N*		4.3
3.0*N*		7.2
U 50 gm/l HNO_3 at 0.6*N*		2.1
1.5*N*		2.4
3.0*N*		2.7
U 400 gm/l HNO_3 at 0.6*N*		0.38
1.5*N*		0.39
UO_2(II) in 1*M* HNO_3	20% tributyl phosphate (*413*)	5†
6*M* HNO_3		30†
0.1*M* HNO_3 and 6*M* $NaNO_3$		Very high†

* See plutonium table for footnotes.

GROSS FISSION PRODUCTS

Aqueous phase	$10^3 \times$ *distribution coefficient**
2*N* HNO_3, 0 gm/l U	8.5
10 gm/l U	3.1
50 gm/l U	2.1
100 gm/l U	1.7
250 gm/l U	1.6
400 gm/l U	1.6
0.6*N* HNO_3, 100 gm/l U	1.4
3*N* HNO_3, 100 gm/l U	2.6

* Based on β-activity measurements, fuel stored for 6 months after irradiation. Zr-Nb is the sole cause of activity carryover. Coefficients of $1\text{–}2 \times 10^{-3}$ are found for γ-activity. Organic phase is 33% tributyl phosphate.

FISSION PRODUCT NITRATES

*Distribution coefficient**	*Aqueous phase* A	B	C
Zr(IV)	0.01	1.0	1†
Ce(IV)	0.01	0.2	
RuNO‡	10	0.01	Very high
Ce(III)¶	0.04	0.02	10
Y(III)	0.01	0.02	
Nb	0.01	0.01	

* Distribution coefficients are in terms of molalities. Organic is 20% tributyl phosphate. Aqueous is A 1*M* HNO_3, B 6*M* HNO_3, C 0.1*M* HNO_3, plus 6*M* $NaNO_3$. All elements are at tracer concentration except for Ru and Pu, which are 10^{-1}–10^{-3}*M*. Stirring times are ½–5 min.

† Aqueous phase is 1*M* HNO_3, 5*M* $NaNO_3$.

‡ This element was extracted as trinitrate.

¶ For Ce, 50% tributyl phosphate is the organic phase.

PLUTONIUM

Aqueous phase	*Organic phase*	*Distribution coefficient**
Pu(VI) in 1.0*M* $Al(NO_3)_3$¶ at *p*H 0.0	Methyl isobutyl ketone (*824*)	3.2
*p*H 0.5		2.9
*p*H 1.0		2.7
*p*H 1.5		2.5
*p*H 2.0		2.2
*p*H 2.5		2.0
Pu (IV) in same at *p*H 0.0		3.0
*p*H 0.5		0.85
*p*H 1.0		0.27
*p*H 1.5		0.09
*p*H 2.0		0.03
*p*H 2.5		0.01
PuO_2(II) in 1*M* HNO_3	20% tributyl phosphate (*413*)	0.7†
6*M* HNO_3		3†
0.1*M* HNO_3 and 6*M* $NaNO_3$		Very high†
Pu (III) in 1*M* HNO_3		0.015
6*M* HNO_3		0.01
Pu(IV) in 1*M* HNO_3		1.3†
6*M* HNO_3		20†
0.1*M* HNO_3 and 6*M* $NaNO_3$		Very high†
Pu‡ 0.5 gm/l, HNO_3 0.5*N*	33% tributyl phosphate (*549*)	1.05
1.4*N*		2.3
2.0*N*		5.5
2.8*N*		10.0
Pu‡ 0.5 gm/l, 2*N* HNO_3, plus 50 gm/l U		0.21
350 gm/l U		0.18
420 gm/l U		0.10
Pu‡ 0.5 gm/l, HNO_3 1.8*N*, plus $KBrO_3$		4.8
Same, plus also $NaNO_2$		8.0
Pu‡ 0.5 gm/l,** HNO_3 3*N*, plus $KBrO_3$		0.39
Same, plus also $NaNO_2$		0.68

* Coefficients in terms of molalities for (*413*). Conditions not given were unstated, except for temperatures of 18–25° C for (*413*). See (*593, 541*) for additional data.

† Stirring time was ½–5 min. UO_2 and PO_2 were as the nitrates, at a concentration of 10^{-2}–10^{-3}*M*.

‡ Valence of Pu + 4, sometimes + 6.

¶ HNO_3 and NaOH used to adjust *p*H.

** Solution has 50 gm/l U.

Solvent-Extraction Plant Design

Continuous processing is preferred over batch processing because (for short reaction times) smaller holdup volumes are involved for the same through-put. Smaller holdup volumes ease the problems of shielding and criticality hazards (see p. 63 for a discussion of criticality). The usual economic advantages—cheapness of operation, product uniformity, and minimum losses—also favor continuous operation.

Specific criteria. Solvent extraction depends on the greater solubility of a material when a system of two solvents and the material is mixed. The two solvents separate into heavy and light phases. Low mutual solubility of the solvents is desirable to obtain favorable extraction and avoid solvent losses.

A high distribution coefficient is not necessarily advantageous in continuous countercurrent extraction, because in packed columns there is a limit to the flow ratio of the phases that can be efficiently used. Separating power of a solvent for two materials is the ratio of distribution coefficients. For some solvents used in separating U from fission products, separating power (Fig. 12) is from 500–5,000.

Separating power will be affected by temperature, solute concentration, relative volumes of the phases, and impurities. Process variables that affect performance of continuous-extraction units are the same three, plus number of stages and location of feed point. These are not independent variables, but are factors to consider when selecting equipment. High solute concentrations can cause trouble with solids depositing and even forming a complete block. Or a viscous third phase can form, interfering with liquid flow and causing premature flooding. These "accumulation" phenomena frequently limit the usable phase-flow-rate ratio.

Wastes from some solvent-extraction processes are free of salts other than those formed by dissolving the fuel elements in HNO_3. Thus the effluent can be evaporated to reduce the volume and save on storage costs. Often 50–100-fold reductions are used.

Plant design criteria. Radioactivity imposes restrictions on equipment selection. Extreme reliability is most desirable, and usually comes with simplicity. Standard equipment is used wherever possible. Very special preinstallation inspection and performance specifications are needed. Equipment must be leakproof, so valves are avoided wherever possible. Welded joints are preferred to flanges.

Methods of elevating solutions can be placed in the following order of reliability: vacuum transfer, air-jet, direct pumping. Low holdup of process solutions should be emphasized in selecting equipment.

Plant is usually designed so it can be drained and decontaminated. Plant servicing philosopy can be any or all of: contact servicing, contact servicing after decontamination, remote servicing by repair, remote servicing by replacement, no servicing—use built-in spares. Plant layout and plant location are guided by the unique features of high radiation and waste disposal. There are strong economic arguments in favor of a single processing station to serve several reactors.

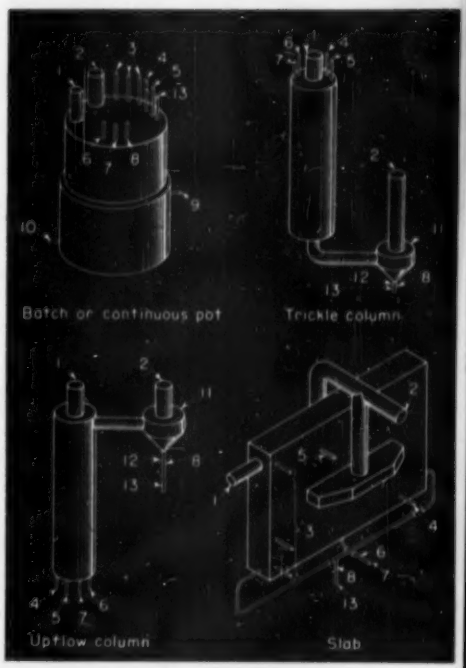


FIG. 11. First step in chemical processing of fuels, dissolving cladding and fuel, is done in one of these (*547*). Because of criticality, capacity of pot is small for enriched fuels. Column dissolvers are best for enriched fuels. In trickle-type all phases flow down at flow rates of $1/10^3/10^3$, solid/liquid/gas. In upflow units, solids go down, other phases up. Slab dissolvers are used for enriched fuels with shapes too unwieldly for column dissolver. Legend: 1-fuel loading, 2-off-gas line, 3-liquid-level, specific-gravity probe, 4-thermocouple well, 5-pressure tap, 6-feed inlet, 7-condensate return line, 8-sampler, 9-pipe for jacket steam and water, 10-condensate drain, 11-cyclone, 12-product-specific-gravity determiner, 13-product transfer line

Typical Plant Processes

Plant Safety

With attention focussed on radioactive materials, there is a danger of neglecting inactive chemicals that may bring inflammability, carcinogenic, and other dangers to plant operators. The possibility of long-term, accumulative toxic effects from new or even common chemicals must be watched for in new processes (414).

Based on part of the data (left), these conclusions (*349*) are drawn: (*a*) distribution coefficients decrease in this order—U, Pu, fission products, with very low values for fission products; (*b*) presence of U strongly lowers extraction of Pu from fission products; (*c*) increasing acidity aids extraction of all elements, the influence being less and less sensitive as U concentration increases; (*d*) an oxidant, such as NO_2^- favors Pu extraction; and (*e*) except for very strong acid or in the presence of salting-out agents Pu(III) is not extracted. The following cycle (*349*) is based on these observations.

French Practice

1. Fuel is decladded (probably mechanically) or with NaOH for Al or HNO_3 for Mg cans. Fuel is dissolved in 12*N* HNO_3 at boiling tem-

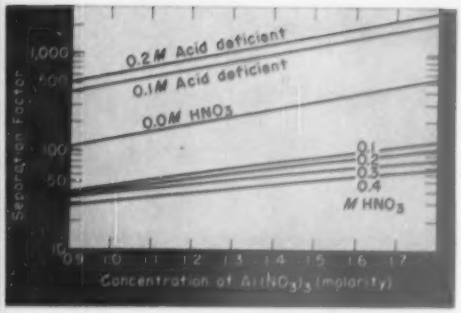


FIG. 12. Separation between U and fission products increases with molarity of salting-out agent, and acid-deficient conditions (*541*)

perature of 110° C. Oxygen is added to reclaim the nitrous vapors, cutting acid consumption to 2 moles/mole U. Solution is adjusted to 1.65*M* U, 1.7*N* HNO_3.

2. Solvent of 40% tributyl phosphate in paraffin-type hydrocarbon contacts the solution in a battery of six countercurrent mixer-settlers. Solvent should be very pure, especially free of dibutyl acids, which extract Zr and Nb. Solvent flow rate is 2.8 times aqueous flow. Outgoing solvent has all the U and 99.8% of the Pu. About 5% of the β-activity and 1% of the γ-activity is entrained with the organic. Outgoing solvent is washed in two supplementary mixer-settlers, with 3–4*N* HNO_3. This back-extracts 70% of the fission products that were carried over. Wash acid is reintroduced. Final product decontamination is 10^6–10^7.

3. Pu is back-extracted into aqueous by reducing it to valence 3 in four mixer-settlers. For low Pu concentrations 0.2*N* HNO_3 and 0.3*N* hydrazine is used. Another reducing agent used contains 0.0025*M* ferrous sulfamate plus the other two reagents. Water phase is ⅓ of solvent phase. Mixing time is 3 min, settling time is 20 min.

4. Aqueous phase Pu is precipitated with NaOH at 8 or 9 *p*H. Precipitate is redissolved in HCl or HNO_3, depending on whether solvent extraction or oxalate precipitation is used for further purification. Conditions for precipitation are 0.7*N* HNO_3 or HCl, and 2.2 moles oxalic acid/mole Pu. Zr and Nb oxalates do not precipitate. In addition, anion-exchange resins can be used for the final purification. Tetravalent Pu is strongly fixed on quartenary ammonium resins with two methyl groups and two 8, 10, or 12 carbon-atom groups, such as didecyl dimethyl ammonium nitrate. Pu then can be eluted with aromatic hydrocarbons or chlorinated solvents. This operation separates U and Pu. Pu yield is 97%.

5. Uranium is reextracted using 2 volumes of water/volume of solvent, in six mixer-settlers. U is finally recovered by oxalate precipitation (1.2–1.5 moles/mole U). The oxalate is heated to 125° C, then calcined at 310° in a reducing atmosphere, to get UO_2. Yield of U is 99%.

6. Fission-product solution is concentrated by evaporation, reducing volume 100-fold. Using formaldehyde in the evaporator reduces the HNO_3 concentration, easing corrosion.

USA Practice

1. Fuel and cladding are dissolved (Fig. 11) in 60% HNO_3 (>11.7*M*) at 105° C, using 1–2% of mercuric nitrate as catalyst to aid Al dissolution for MTR-type fuel elements (*541*). Alternative is to use 3–8*M* NaOH to dissolve Al cladding, where silicon bonding has been used (*539*), and then HNO_3 at 105° C to dissolve U. Acid consumption is 4.5–4.8 moles/mole U. HNO_2 is added to ensure Pu(IV). Feed solution is filtered through a 10-μ pore-size sintered-stainless-steel filter.

2. U, Pu, and fission products are separated by a series of extraction and stripping steps, carried out in continuous-flow packed or pulsed columns. For flowsheets see NU, Aug. '55, p. 22 and March '55, p. 18. Methyl isobutyl ketone or tributyl phosphate is the organic. Aqueous phase used in the first extraction step is acidic in HNO_3 and contains $Al(NO_3)_3$ as salting-out agent. Feed enters the column at about midpoint. Pu is not reclaimed from MTR fuel (*541*). Below feed point, U and some fission products are extracted into organic; above feed point, the organic phase is scrubbed with a dilute $Al(NO_3)_3$ water solution. Although U and fission products are in general affected the same way by changing conditions, the separation factor (ratio of distribution coefficients) changes enough (Fig. 12) to allow excellent separation to be made.

3. Organic phase with U and Pu goes to a partitioning column where Pu is reduced to +3, in presence of dilute HNO_3. Under these conditions distribution coefficient of Pu is 0.02 (*539*). A similar operation with a reducing agent added has a distribution coefficient of 6×10^{-4} for Pu. Aqueous Pu solution, if Pu recovery is desired, flows to a second cycle. $NaNO_2$ (in 3-fold excess) oxidizes Pu to +4, and after adjustment with HNO_3, another extraction into organic is carried out. This step leaves the rare earths in aqueous solution. HNO_3 wash separates the fission products that have carried over. Pu is then stripped into aqueous by another reduction. Ion exchange makes the final concentration.

4. The organic U solution from the partitioning column is fed to the bottom of a stripping column where dilute HNO_3 enters. Now the aqueous U solution is treated by another cycle of organic extraction and back-extraction. Concentration by continuous evaporation and addition of HNO_3 produces the proper feed conditions. An alternative procedure is batch evaporation followed by adding $Al(NO_3)_3$ and NH_4. Final concentration and purification is by either filtering through a silica-gel packed column, and elution with dilute oxalic, or by another solvent-extraction cycle. At MTR, U yield is 99.85%.

Processing Problems

Tomlinson (*824*) details some problems in operating solvent-extraction plants. Among them are transition of Ru from extractable to nonextractable forms (see *437*, *670*). Another is hydrolysis of elements. Reversible hydrolysis forms less extractable com-

Cooling Period Before Processing

Enriched-uranium fuel elements are stored under water before processing to allow for fission-product activity decay. Storage period usually is determined by the quantity of U^{237} (6.9 day) produced during irradiation:

$$U^{235}(n,\gamma)U^{236}(n,\gamma)U^{237} \rightarrow Np^{237}$$

U^{237} will be present in product uranium. It must be allowed to decay before low-activity uranium can be produced. This decay also allows time for fission-product activity-level reduction of 10^4. At decay periods >100 days, the active fission gases I^{131} and Xe^{133} have decayed, and other fission products such as Ba^{140} (12.8d)-La^{140}(40.0h) have been reduced. La^{140} decay is desirable before processing, because this isotope emits a 2.1-Mev γ which controls chemical-plant shield thickness for decay periods <90 days.

plexes, but can be controlled by *p*H adjustments.

TBP hydrolysis. Irreversible hydrolysis of tributyl phosphate forms dibutyl phosphate, which forms tight complexes with U, Pu, and Zr, increasing product loss of the first two and increasing contamination of recovered product. Any monobutyl phosphate forms precipitates with U and Pu, increasing loss.

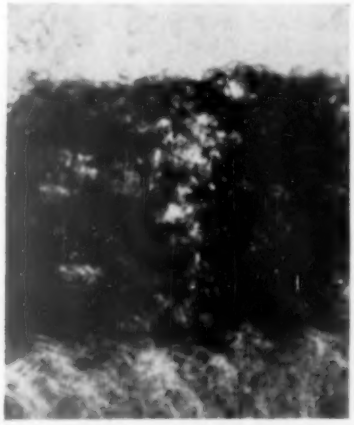
Presence of solids and emulsifiers. Solids less than a few μ in diameter are carried on the interface, by the dispersed phase. If, as silica does with Zr and Nb, they adsorb radioactive ions, these solids limit the decontamination attainable.

Emulsifying agents cause solvent entrainment by the aqueous phase, and reduce decontamination. In severe cases, emulsification can prevent countercurrent flow. Gelatinous materials, MBP precipitates, colloidal silica or alumina are examples of emulsifiers that must be excluded.

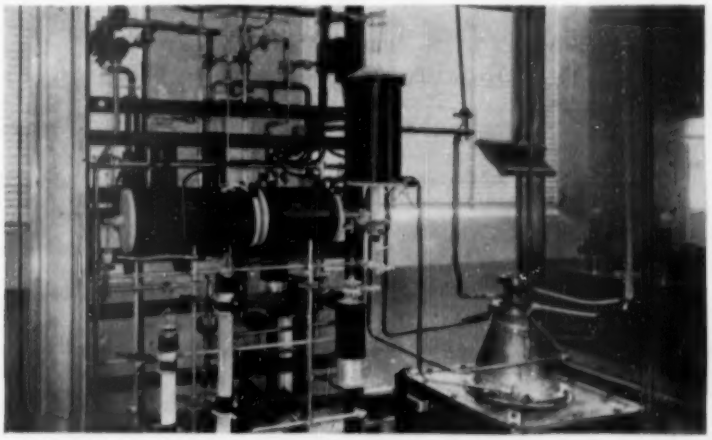
Piloting the process. Many plant operating problems do not become apparent until many days of operation. To insure that the process will produce satisfactorily, a pilot plant should be operated before start-up of the production facility, preferably before its design and construction. Cost is usually recovered via decreased plant modifications and increased productivity from the start of operations.

Other Extraction Agents

Successful tests on extracting Pu from U use chloroform as organic phase, and tetrabutyl ammonium nitrate, cupferron or neocupferron, as complexing agents (*785*). Ammonium nitrate also was added.



U INGOT obtained by melting in MgO crucible for 2 hr at 1200° C in He shows dark oxide layer above U metal (*544*)



MELTING FURNACE for oxidative slagging of irradiated fuels shows inert-atmosphere piping system. Most fission products enter the slag in this process being developed for treating reactor cores (see p. 59)

Safe Criticality Limits for Pu Solutions

A system is safe if it falls within any one of the limits given in the table. Thus, less than 464 gm of Pu is safe in any vessel at any concentration. Values in the table are for Pu solutions at room temperature, in a homogeneous solution with H density equal to that of pure water. For the reflected case, an infinite thickness of water is postuated. Usual practice is to apply a ⅔ safety factor to the figures, and also use reflected-system-without-safety-factor figures for substantially unreflected systems. To make the plant, rather than its operation, safe, preference is given to using vessel dimensions as safety basis. A sphere is the most dangerous shape, so any vessel with less than safe-sphere volume will be safe also (*454*).

System	*Quantity restricted*	*Calculated limits*		*Limits with ⅔ safety factor*	
		H_2O-reflected	*Bare*	*H_2O-reflected*	*Bare*
Any vessel, any dimensions	Concentration	6.6	6.6	4.4	4.4 gm/l
Same	Mass	464	1,244	309	828 gm
Any tube, any cross section	Mass/unit length	10.0	22.4	6.6	149 gm/cm
Any slab-shaped vessel, any thickness	Mass/unit area	0.39	0.24	0.26	0.16 gm/cm²
Any vessel, any mass of Pu	Volume	4.5	20.3	3.0	13.4 liters
Any tube, any mass	Cross section	98	201	80	164 cm²
Spherical vessel, any mass	Diameter	20.5	33.8	17.9	29.5 cm
Cylindrical vessel, any mass	Diameter	12.5	25.6	10.2	20.9 cm
Slab-shaped vessel, any mass	Thickness	4.0	17.6	2.7	11.7 cm

reactor materials

Heavy-water production—beryllium production and fabrication—aluminum and zirconium corrosion—solid and liquid radiation damage

Heavy-Water Production

TABLE 1—By-Product Heavy-Water Proceses (*829*)

Process	*Ammonia-synthesis gas distillation*	*Water distillation with low-pressure by-product steam*	*Water electrolysis*	*Water electrolysis plus steam-hydrogen exchange*
Main product/lb D_2O	H_2:162 MCF,* or NH_3:2.2T†	10,800 kwh	H_2:390 MCF, or NH_3:5.2T	H_2:224 MCF, or NH_3:3.0T
Separation factor	1.39	1.05	7	7, 2.8
D_2 recovery	86%	3.9%	33%	61%
Moles vapor processed/mole D_2O	34,700	198,000	21,400	18,200
Energy input/lb D_2O	1,690 kwh	178,000 lb 22 psia steam	57,800 kwh	31,800 kwh
Unit costs				
Plant ($/lb D_2O/yr)	170	155	?	178
Operation ($/lb D_2O)	16	34	?	38
Total, plant charged at 10%/yr ($/lb D_2O)	33	50	13	56
Cost assumptions	Natural gas at 10¢/MCF	High-pressure steam at 40¢/1,000 lb	H_2 made for 50¢/MCF	
Engineering problems	Handling large amounts liquid H_2	None	Development of electrolytic cell with high separation factor	

* MCF—1,000 ft³ of gas at 60° F and 1 atm.
† T—1 short ton (2,000 lb).

Selecting the best heavy-water production process (*829*)* depends fundamentally on the production rate required and on availability of low-cost energy sources or favorable feed situations. Costs will be prohibitive except on a long-term, large-scale basis—that is, at least several tons per year for 10 years or more.

The production processes may be either single or dual purpose. In the latter, heavy water is the by-product of another operation that carries the major part of the cost. By-product processes have an economic advantage, but their rate of heavy-water production is limited by the rate at which the main product is made so that none can produce enough heavy water to support a large nuclear-power industry. To produce more than 40 tons/yr D_2O (one reactor charge may be 200 tons), one or more of the single-purpose processes is required. Because they use water as the primary feed, their potential capacity is unlimited.

By-product processes are summarized in Table 1. The following is additional information on each process.

* References to UN papers on p. 94.

Ammonia-synthesis gas distillation. Plants must handle unprecedented 20-million ft³ or more of hydrogen a day. Cheapest way of producing D_2O at 20–40-ton rate per year (upper limit set by maximum capacity now available at any one site).

Water distillation with low-pressure by-product steam. Can be used where fuel is cheap and large block of electric power needed. 100,000-kw steam power plant would produce 40 tons/yr D_2O. Better tower internals may reduce cost.

Water electrolysis. Few places in world where electrolytic hydrogen can be made for cost given. Capacity is small. By increasing separation factor to 7, only 5 tons/yr could be produced at plant making 500,000 ft³/hr of hydrogen.

Water electrolysis plus steam-hydrogen exchange. Exchange facilities would almost double electrolysis production. Cost would probably increase. Electrolytic-hydrogen supply would limit capacity. Catalyst not subject to gas poisoning would provide larger hydrogen volumes and reduce cost.

Single-purpose processes are summarized in Table 2. Cost data is not given because of inadequate information from case studies. Additional material on each includes the following:

Hydrogen-fed D_2O distillation by steam exchange. Costs more than ammonia-synthesis gas distillation. Addition of high-temperature exchange equipment adds to plant cost. Hydrogen entering low-temperature plant carries 0.01% instead of 0.013% D in ammonia synthesis gas. For given production rate, the low-temperature plant is bigger and power demand greater.

Water distillation with vapor recompression. Costs more than low-pressure by-product steam. Power demand of 10,000 kw/lb D_2O excludes process unless tower packing with low-pressure drop per theoretical plate can be developed.

Dual-temperature H_2S water exchange. Advantages lie in handling smaller amounts of gas and lower energy demand. Corrosion problem not met in more complex dual-temperature, ammonia-hydrogen exchange (coupled with water-ammonia exchange).

Dual-temperature steam-hydrogen exchange. With one equilibrium contact per stage at high temperature, uses too much plant for low-cost production. Energy demand is very low. Simpler equipment with greater consumption might improve process. Cheap catalyst permitting conventional absorption towers would make this most economical of all single-purpose processes.

TABLE 2—Single-Purpose Heavy-Water Processes (*829*)

Process	*Hydrogen-fed D_2O distillation by steam exchange*	*Water distillation with vapor recompression*	*Dual-temperature H_2S water exchange*	*Dual-temperature steam-hydrogen exchange (single hot contact per stage)*
Separation factor	1.39	1.05	1.22	<2.15
D_2 recovery	13%	3.9%	<18%	<32%
Moles vapor processed/mole D_2O produced	108,000	198,000	>72,000	>140,000
Energy input/lb D_2O	~2,300 kwh, plus heat for exchange system	~10,000 kwh	Small	>7,600,000 Btu
Engineering problems	Large amounts liquid H_2; preventing steam-hydrogen reaction in heat exchanger	None	Corrosion	Preventing steam-hydrogen reaction in heat exchanger

Beryllium Production

Beryllium oxide is about as good as Be metal as a neutron moderator (*820*). It is chemically inert and has a high melting point. However, it is brittle and not easily adapted to engineering uses. No USA reactors use BeO.

Thermal-rupture resistance can be improved by small additions of Al_2O_3 or ZrO_2 and by controlled porosity. Unfortunately, reactor builders want as high density as possible. Although the thermal conductivity of BeO is high, it decreases with increasing temperature (0.34 watts/cm/cm²/° C at 540° C, 0.17 at 980° C). Fast-neutron irradiation decreases conductivity (decreased 40% by 1.3×10^{20} nvt). BeO could be used in a low-power reactor; further improvement in thermal-rupture behavior is required for high-performance-reactor use.

Be production. The bulk of pure Be metal in the USA is made by magnesium reduction of the fluoride and vacuum casting. The analysis of the 99.5% Be vacuum casting is (in ppm): 400 C, 1,100 Fe, 70 Mn, 90 Cr, 130 Ni, 500 Mg, 600 Al, 1.2 B, 0.5 Cd, 1 Li, 5 Ag, 2 Co.

Extremely pure Be has been produced in the USA by electrolysis of $BeCl_2$. However, the Be powder produced contains enough slag to cause fabrication difficulties.

The USSR (*633*) reports pure Be production by magnesium reduction of the chloride followed by vacuum distillation. Impurities are (in % Be content): 0.005 Fe, 0.003 Al, 0.004 Cu, 0.003 Ni, 0.005 Cr, and 0.002 Mn.

BeO production. A USSR process is based on distilling (360–400° C) the basic acetate [$BeO{\cdot}Be_3(CH_3COO)_6$] prepared by treating the hydroxide with acetic acid. After two distillations, the basic acetate is decomposed at 600–700° C to form the oxide as a highly dispersed powder. Said to meet purity requirements for reactor use, the BeO contains the impurities (% Be content): 0.001–0.003 Fe, 0.003–0.007 Al, 0.010–

0.020 Si, <0.003 Mn, <0.0005 Cu, <0.002 Ni, and 10^{-4} B.

BeO is commercially produced in the USA by calcining $Be(OH)_2$ obtained with the fluoride and sulfate processes. In both cases the grade of BeO is of sufficient purity for production of beryllium copper, but unsuitable for refractory use mainly because of its sodium content.

To produce a grade suitable for nuclear use, the $Be(OH)_2$ is dissolved in sulfuric acid, and ammonium sulfate is added to separate the remaining Al. The $BeSO_4$ is converted to BeO by calcining at 1,150° C. The pure oxide is fluffy and contains only small amounts of alkalies.

Be metal as-cast is exceedingly brittle because of coarse grain size. Electrolytic flake further purified by vacuum melting and distilled Be shows no improvement in ductility (*820*); it appears that further purification will improve it. Although oxygen may cause brittleness, vacuum-melted Be appears to contain the least and is the least ductile. Alloying with the few elements having appreciable solubility in Be appears to make the metal more brittle.

Powder metallurgy. Despite dry-atmosphere milling, the BeO content increases from 0.1–0.2% in the vacuum-cast metal to 0.7–1% in the finished −200 mesh powder (*820*). Despite the increasing oxide content, fine-grain Be requires less pressure for densification than coarse-grained metal and shows superior mechanical properties at room temperature. However, above 500° C, coarse particles give better mechanical properties. As a compromise, commercial −200 mesh powder (2–200 microns, 20-micron average) has been chosen. In this form, powder can be stored in glass bottles for long periods without deterioration; powders made by other methods—containing chloride, reduced by thermic or electrolytic methods and containing slag—suffer deterioration.

High heat resistance, which depends largely on density, is important for reactor use (*633*). Articles for long-term, high-temperature service can be made by hot-pressing the powder under vacuum at 1,100° C. Thermal stresses are removed by annealing under vacuum at 1,100° C and slow cooling. For high-density (1.84–1.85 gm/cm³) articles to be used at lower temperatures, powder or chips are hot-pressed in a heat-resistant-alloy open die at 600° C and 5 tons/cm².

Rods and tubes can be made by hot extrusion of powder-metallurgy ingots. Porosity is <0.5–1%. After annealing, Brinell values are 90–115 kg/mm² and tensile strength 50–70 kg/mm²; that of extruded bars is 65–90 kg/mm².

Aluminum Corrosion

Above about 200° C, aqueous corrosion of aluminum apparently occurs in a manner opposite to that at room temperature. The pure metal rapidly disintegrates while alloys show better resistance. To understand and control this behavior, it is necessary to determine the effects of surface treatment, solution inhibitors, and *p*H.

Tests at 230° C. Of commercial alloys tested in distilled water with O_2 and N_2 atmospheres (*880*, Norway), 2S Al appeared the most promising followed by Polital (1% Si, 0.7% Mn, 0.8% Mg) and Pantal (1% Mg, 1% Si, 1% Mn). Superpurity Al and Reflektal (0.5% Mg) disintegrated.

FUEL-ELEMENT failure apparently due to pitting-type corrosion in aluminum jacket. Natural-uranium element (1.1-in. diameter, 0.035-in. Al) underwent 6 yr operation in CP-3; failure first detected by D_2O-moderator contamination (*861*)

JEEP ELEMENT (1-in. diameter, natural uranium, aluminum canned) removed due to blistering. Blister, 3 mm high, was scratched during removal from reactor. Formation attributed to moisture inside canning resulting in uranium oxide or hydride (*888*)

Alloys of Si, Fe, Si-Fe, and Cr added to superpurity Al were tested. For effectiveness, the Si must be dispersed by annealing at over 450° C; 1% Si is extremely effective. Al-Si-Fe and Al-Fe were also effective; because the iron contained 0.008% Si, it is not known whether the Fe was the active agent.

Keeping the solution *p*H at 3.5–5.0 with H_3PO_4 and H_2SO_4 appears to retard corrosion. However, with *p*H below 3.0, serious acid attack may result.

Electropolishing, abrading, and pre-irradiation treatment in inhibitor at 230° C have little effect. Anodized samples, sealed in chromate and silicate, maintain a protective layer during corrosion. No improvement is obtained with enamel, chromium-plate, or silicon-vapor coatings.

Several corrosion inhibitors were added to distilled water at 230° C to test the effect on 2S Al. Very good inhibition is obtained with 1% $K_2Cr_2O_7$ (some pitting), 1:1 dichromate-Na silicate (no pitting), and 1% pyrogallol (some decomposition); fairly good with 1% $K_2Cr_2O_7$ neutralized with NaOH, 0.1% K_2CrO_4, saturated As_2O_3 (extensive pitting), and 1% hydrochinon; bad with 1% pyridine; and very bad with 1% $Al(SO_4)_3$, 1% Na silicate, and 1% $NaMoO_4$ (black, lot of pitting).

Tests at 315° C. In tests made at higher temperatures (*535*, USA), a more violent attack on 2S Al is described in which blisters form, the surface roughens, and the metal is penetrated more rapidly, producing mixtures of metal and oxide. Samples have disintegrated in distilled water in 4 hr. Other alloys, e.g., B195, 356, 142, Al-Si eutectic, 355, 122, Al32, 248, and 188, show greater resistance.

This blister attack is believed caused by pockets formed in the metal by corrosion-produced hydrogen that eventually burst, leading to further attack. Aluminum impervious to H_2 diffusion, e.g., vacuum-cast, resists well, but loses its resistance when worked.

Another effective measure is to put certain reducible metal ions in solution

(minimum corrosion was obtained with sulfates of Co, Cd, Ni, and Cu). The metal deposits on cathodic sites on the Al and liberates hydrogen produced in the cathodic reaction. Complete protection at 275° C is provided by 5 ppm by weight of Ni^{++} in water. It is necessary to add acid to the solution to prevent formation and deposition of basic salts; although excellent protection is provided 2S Al, stainless steel corrodes rapidly in this environment. Electroless nickel plate (0.0007 cm) on 2S Al surface has excellent resistance.

Al-0.5% Ni alloy gives good protection at 315° and 350° C. Silicon and iron added to Al-1%Ni alloy indicate that more than 0.1% Si causes rapid corrosion failure at 350° C, while 0.2–1% Fe makes highly resistant alloys.

Film protection of surface. The passivity of Al depends on the formation and maintenance of protective films, particularly Al_2O_3 as opposed to hydroxide or oxyhydroxide (*411*).

In water, the film formed up to 70° C is hydroxide and up to at least 300° C the oxyhydroxide. "Anodizing" complex anions containing oxygen, e.g., chromate and silicate, are necessary for forming oxide. The absence of these anions in purified water may lead to extreme corrosiveness. These inhibitors may act by adsorbing at the anodic surface, excluding OH^- from the reaction, and supplying O^{2-} to the film.

Pitting is the most important aspect of aluminum corrosion in reactors. Contact can be avoided with noble conductors such as copper and graphite, but it is difficult to avoid trace quantities of copper ions in solution; 0.1 ppm promotes pitting. Anodic inhibitors are not always effective. In that case, the noble-metal ions must be eliminated from solution.

Zirconium-Alloy Corrosion*

Alloying element	*Concentration (weight %)*	*Effect*
Binary alloying elements		
Sn	0.5–3.0 for 200–800 ppm N	Counteracts nitrogen impurity; optimum conc. depends on N content. Zr-Sn-N not equal to pure Zr
Fe, Ni, Cr	0.1–0.5	Improves corrosion resistance
	>0.5	Improves resistance in steam (400–816° C); harmful in hot water
Mo	0–10	Harmful
Nb	0–10	Tendency to counteract nitrogen; much less effective than tin
	10–30	Harmful
Sb	0–3	Tends to counteract nitrogen; much less effective than tin
U	0–100	Corrosion resistance decreases with increasing U content
Si	0–1	Harmful
Ti	0.1–100	Corrosion resistance extremely poor up to at least 20%
Ta	0.05–10.0	Corrosion resistance decreased
Hf	0–2	No effect
Co	0–1.2	Corrosion resistance decreased slightly
W	0.7	Corrosion resistance decreased
Cu	0–1.5	Corrosion resistance decreased
Mn	0–5	Corrosion resistance decreased
Al, Cd, Ce, Ca, Mg, Nd, Pb, Zn, La, Ga, V, Be		Harmful
Ternary alloying elements		
Sn + Fe, Ni, or Cr	1.8 0.25	Marked improvement over Zr-1.8% Sn
Sn + Mn, W, Mo, Ta, Ti, Cu, Nb, or O		Less effective
Zircaloy-2 (1.5% Sn, 0.12% Fe, 0.10% Cr, 0.05% Ni)		Good corrosion resistance
Zircaloy-2 + Al, Pb, Si, Cu, or Mn		Not effective; harmful at certain conc.
Zircaloy-2 + Nb, Mo, Ti, or Be		No effect
Sn +Fe or Fe-Ni	0.25–0.50 0.25–0.40	Most promising

* Aqueous corrosion at high temperature (*537*).

Radiation Damage

At the present imperfect status of radiation-damage knowledge (*744*), it cannot be reduced to an engineering calculation. To learn more about basic mechanisms, more materials-testing reactors and hot laboratories are needed. Current research programs in the USA are investigating:

1. Mechanisms and kinds of defects in solids involved in neutron and fission-fragment interactions.

2. Effect of these defects on reactions such as diffusion, precipitation, nucleation, phase transformations.

3. Materials most suitable for immediate reactor application by engineering property measurements under operating conditions.

4. Effect of irradiation on combinations of components constituting parts of a proposed reactor system.

Outside of these programs, there are many other unanswered problems.

The number, temperature stability, and relative importance of the several types of bombardment defects is not known. Do cyclotron and other charged-particle machines give information pertinent to reactor problems? Are present values of energy required to displace an atom from its equilibrium position as determined by electron bombardment correct or high

by a factor of 2–5? Neutron-flux distribution is not well known. Mechanism of radiation effect on many mechanical properties is not known. Knowledge of the solid state in the absence of irradiation is far from complete.

The following is some information that we do have on radiation damage of solid reactor materials.

Metals

Physical changes due to exposure depend on atomic number (effect on number and types of defects produced), melting point (defect migration and recovery at given exposure temp.), and crystallographic structure (larger changes in tensile properties of body-centered-cubic metals than of face-centered-cubic and hexagonal-close-packed metals (*747*).

Reactor or fast-charged-particle exposure produces increased hardness and strength and decreased ductility (*747*). The percentage change in metal properties due to irradiation is appreciably smaller for cold-worked than for annealed samples (*747*).

In the case of metals exhibiting brittleness below a certain temperature and ductility above that temperature, irradiation increases the brittle-ductile transition temperature. Irradiation decreases the energy required to fracture material in the ductile region. It appears that irradiation does not induce brittle behavior in metals not exhibiting it in the unirradiated state (*747*).

Irradiation has no effect on macroscopic diffusion rate (*747*).

Aluminum. The alloys 2S, 52S, 61S, A 54S show increased yield stress and decreased ductility after 10^{21} n/cm² (*744*). Cold-worked 2S exposed to 2×10^{20} n/cm² (>0.5 Mev) at 30° C has lattice distortion (in addition to that due to cold working) and increased hardness (+15%), ultimate tensile strength (+26%), and yield strength (+22%) (see Table 3) (*747*). No hardening is caused by neutron irradiation at 250–300° C (this is above the Al recrystallization temperature) (see Table 4) (*680*). Cyclotron irradiation has no effect on the creep rate (*747*); neutron irradiation causes no change in the creep rate of polycrystalline Al in the secondary creep range (*444*).

TABLE 3—Fast-Neutron Effect (30° C Exposure) on Tensile Properties of Cold-Worked Al (*747*)

Exposure, n/cm² (>0.5 Mev)	*Initial sample condition*	*Property change (%)*		
		Offset yield strength	*Ultimate strength*	*Per cent elongation*
7.5×10^{19}	Annealed	+100	+60	−67
	Half-hard	+ 10	+ 6	0
	Extra-hard	Very small		
15×10^{19}	Annealed	+100	+70	−67
	Half-hard	+ 10	+ 6	0
	Extra-hard	Very small		

Beryllium. The radiation stability is excellent (*744*). Dose of 3×10^{20} n/cm² (>1.0 Mev), 30° C, increased hardness (+27%); 5×10^{18} n/cm² (>0.5 Mev), 30° C, caused no significant changes in length, density, hardness, modulus, decrement, electrical resistivity, or thermal conductivity (*747*).

Cb, Ti, Ta, W. At 10^{19} n/cm², Ti and Ta have no density or dimensional changes, W has 0.2% density reduction, Cb, Ti, and Ta-75%W give Brinell hardness increases of +35; Ta and W (99+% pure) show no change (*744*).

Cadmium. Alpha bombardment causes no change in single-crystal creep rate at 5×10^{9} α/cm²/sec (*444*).

Copper. Hardening increases under n-irradiation, showing increased grain size (see Table 4) (*680*). Single-crystal wire shows marked increase in critical shear stress with low doses (*444*). Al, Zn, Ga, Ge, As, and Mn alloys have an anomalous electrical resistivity decrease of 1–3%. Hardness and strength decrease (*744*).

Iron. Neutron exposure causes marked hardening (see Table 4) (*680*).

Molybdenum. Young's modulus has no increase up to 5×10^{18} n/cm² (*444*). With a dose of 10^{20} n/cm² (>1.0 Mev) at 80° C, tensile strength and yield point increase +50%; elongation to rupture decreases from 40% preirradiation to zero post-irradiation; and reduction in area decreases from 65% preirradiation to zero postirradiation. With a dose of 10^{12} n/cm² (>1 Mev) at 80° C, the brittle-ductile transition temperature increases from −30° to 70° C (*747*).

Nickel. There is no increase in thermal conductivity in in-pile measurements (*744*); there is hardening and increased grain size (see Table 4) (*680*). With a dose of 2×10^{20} n/cm² (>0.5 Mev) at 180° C, there is increased hardness (+120%) and ultimate tensile strength (+50%) and decreased elongation rupture (−45%); annealing data is given in Table 5 (*747*). With dose of 3×10^{11} n/cm²/sec (>0.5 Mev), creep rate shows little change at 700° C test with a beam loaded with maximum fiber stress of 141 kg/cm²; with a 6×10^{12} n/cm²/sec

TABLE 4—Per Cent Change in Mechanical Properties of Metals after Irradiation (*680*)*

Material	*Ultimate tensile strength*	*Relative elongation*	*True tensile strength*	*Reduction of area*	*Impact strength*	*Microhardness*
Copper	+11	−40	− 9	− 4	−59	+28
Nickel	+56	−55	− 5	− 6	−23	+82
Iron (Armco)	+61	−67	+ 8	−15	−90	+88
Aluminum	+ 3	− 5	−24	− 3	+62	−12

* Irradiation: 250–300° C, 1.1–1.4 × 10^{20} nvt. Annealing: Cu, Fe—700° C; Ni—800° C; Al—400° C.

TABLE 5—Annealing of Hardness Changes in Irradiated Nickel (747)

Annealing temp. (° C)	Annealing time (days)	Rockwell "B" hardness
As irradiated		96
200	1	96
	8	97
400	1	94
	8	78
600	1	48
	8	46

TABLE 6—Effect of Exposure on Zr Creep Rate at 260° C (747)

Stress (kg/cm²)	Rate before exposure (10^{-4}%/hr)	Rate during exposure (10^{-4}%/hr)	Rate after exposure (10^{-4}%/hr)	Ratio of creep rate with pile off to that with pile on (%)
1,265	1.0	<0.045		2,000
2,530		0.23	4.6 (3 hr after)	2,000
1,265	2.8	<0.22		1,300
1,090	2.0	<0.20		1,000
1,300		2.0	13.0 (30 hr after)	650

TABLE 7—Per Cent Change in Mechanical Properties of Steel after Irradiation at 80° C (747)

Material*	Integrated flux (nvt)	Ultimate strength	Yield strength	Relative elongation	Impact strength	Microhardness
1	2.43×10^{20}	+29	+192	−48	−65	+53
2	3.2×10^{20}	+10	+116	−45		+66
3	2.43×10^{20}	+1.5	+148	−26	−55	+66
4	3.2×10^{20}	+21.0	+ 96	−36.5		+80
5	2.43×10^{20}	+27.0	+ 58	−46.5	−43.3	
6	2.43×10^{20}	+44	+116	−97.0		+26

* 1, 2—stainless steel quenched from 1,100° C (18% Cr, 9% Ni, 0.6% Ti, 0.1% C); 3—quenched from 1,100° C (18% Cr, 9% Ni 0.06% C); 4—quenched from 1,100° C (20% Cr, 14% Ni, 2.5% Si); 5—normalized at 850° C (1.0% Cr, 0.25% V, 0.5% C); 6—stainless steel (13% Cr, 0.5% Mo, 0.12% C, 1.5% Nb).

(>0.5 Mev) dose, there is no change at 500° C (747).

Zirconium. Annealing for 100 hr at 250° C removes 75% of the tensile-property changes caused by reactor exposure at 50° C; some remains after 100 hr annealing at 300° C; higher annealing temperature is required for comparable recovery of cold-worked samples (747). Irradiation at 50° C increases the energy required to fracture; the increase is particulary noticeable below 150° C (747). In impact tests conducted at −78° C on samples exposed to 3×10^{19} n/cm² (>1 Mev), the energy to fracture increased 200%; subsequent irradiation to 6×10^{19} n/cm² reduces it to the preirradiation value (747). With dose of 2×10^{20} nvt at 180–240° C, a sample annealed at 900° C increased microhardness (+22%) and specific electrical resistivity (+2.2%) (680). Creep shows a marked decrease after 3×10^{12} n/cm²/sec (>0.5 Mev) (see Table 6) (747).

Constantan. There is no effect on creep rate by 3×10^{13} n/cm²/sec (fast neutrons) at 300° C (747).

Inconel. In-pile measurements show no increase in thermal conductivity (744). Creep-rate decrease was observed at low temperature and slight increase at high temperature (temperature at which increase occurs appears to be function of stress (744); after 5×10^{12} n/cm²/sec (>0.1 Mev), there was little change in creep rate at 705° C with beam loading to the maximum fiber stress of 420 kg/cm² (747).

Steel. Pressure-vessel type: impact strength is reduced considerably; the brittle-ductile transition temperature is raised 50–100° C after 10^{19} n/cm²; electrical resistivity and density changes are slight (744). Steels, including industrial austenitic types, show increase in strength and decrease in plasticity (see Tables 7 and 8) (680). Type 347 stainless: 10^{20} n/cm² (>1.0 Mev) at 80° C causes increased hardness (+10%) and electrical resistivity (+2%); fast neutrons cause a slight increase in creep rate above 760° C and decrease below 760° C; hardness does not increase at 760° C irradiation and increases at <760° C irradiation (747). The yield strength of stainless steel at 10^{20} n/cm² is very close to the ultimate strength; no marked change in Young's modulus was observed up to 5×10^{18} n/cm²; stainless steels show an increase in magnetic susceptibility (possible precipitation of ferrite from metastable austenite) (744). Accord-

TABLE 8—Microhardness of Steel Irradiated at 80° C ($1–3 \times 10^{20}$ nvt) (680)

	Microhardness (kg/mm²)			Microhardness (kg/mm²)	
No.*	Before	After	No.*	Before	After
1	151	289	8	160	320
2	237	367	9	177	300
3	211	319	10	148	272
4	237	360	11	159	277
5	269	340	12	223	280
6	122	300	13	151	266
7	150	300	14	293	411

* Composition of steels: 1—15% Cr, 14% Ni, 2% Nb; 2—25% Cr, 11% Ni, 2% Nb; 3—25% Cr, 6% Al with small amounts of other elements; 4, 5—40% Cr, 10% Al with small amounts of other elements; 6—18% Cr, 10% Ni, 2% Mo; 7—18% Cr, 12% Ni, 1.5% Nb; 8—18% Cr, 12% Ni, 2% Mo, 1.5% Nb; 9—28% Cr; 10—Armco; 11—18% Cr, 0.6% Ti, 0.06% C; 12—18% Cr, 9% Ni, 0.6% Ti; 13—0.10% C; 14—12% Cr.

ing to USSR *615*, neutrons do not cause austenite to decompose to form the ferrite; some mechanical properties improve (ultimate strength and hardness increase), but elongation decreases considerably (remains within tolerable limits).

Nonmetals

BeO. Serious decreases in thermal conductivity and compressive strength at room temperature and dimensional changes of the order of 1% are produced after 10^{19} n/cm². It is chemically stable (*744*).

Ceramics. Spinel, HfO_2, mica, ZrO_2, plate, Pyrex, lead glass, and porcelain have substantial decreases in room-temperature thermal conductivity after a fast-neutron exposure of 10^{19} n/cm²; silica glass and single-crystal Al_2O_3 appear unaffected at this dose.

Graphite

The structure and properties of graphite are changed considerably by particle bombardment, but ionizing radiation is nearly ineffective (*751*).

With a 4-cm neutron free mean path, primary displacements are 4 cm apart along the neutron path, very likely causing single vacancies. The energetic primary-displaced atom loses energy principally by excitation and ionization—an occasional elastic collision producing secondary displacements—until its energy is less than a critical value (assumed to be about 10 kev). Most of its energy is then lost by elastic collisions, producing secondary displaced carbon atoms progressively closer together along the path of the primary. Near the end of its range, the primary-displaced atom may displace atoms so close together that they remain as C_2 complexes. The displaced secondary atom has enough recoil energy on the average to make a tertiary displacement. The displaced atoms form predominantly in clusters, consisting mostly of tertiary atoms near the beginning of the range and secondary atoms near the end.

To eject a carbon atom from its lattice position, a collision must transfer to it more than its displacement energy (about 25 ev). Smaller collision-transfer energies excite lattice vibrational modes; this lattice heating becomes more important near the end of the primary range.

The rate of displacement of atoms can be estimated from the theory of disordering of solids if the energy distribution of the damaging particles is known. This leads to an estimated 0.35–0.7 displacement per 10^4 atoms for each megawatt day per adjacent ton of uranium in reactors where most of the irradiations have been performed. (The megawatt day—Mwd—is associated with an integrated slow-neutron flux of 5×10^{17} n/cm² for graphite-moderated reactors.)

In extruded graphite, the crystal planes are oriented parallel to the direction of extrusion, resulting in anisotropic behavior; thus the "cut" of the sample is specified as transverse (perpendicular) or parallel to the long axis of the extruded bar.

In-pile radiation damage often causes severe changes in the over-all physical properties of graphite (*746*). Mechanical properties are improved (hardness and strength increased). Thermal and electrical conductivities decrease; graphite shapes exhibit changes in their gross dimensions; and several other effects, such as storing of potential energy, are encountered. All effects are decreased in magnitude when the graphite temperature during irradiation is increased.

The following physical effects of radiation damage were produced by irradiating extruded graphite samples with fast neutrons mainly at 30° C.

None of the data obtained from Hanford-reactor irradiations indicate any dependence on neutron-flux intensity; if present, it is masked by the effects of small temperature variations in the samples during irradiation and by small differences in the initial properties of presumably duplicate samples. However, recent irradiations in the MTR have produced results in poor agreement with those of Hanford irradiations. Whether they are caused by the higher MTR flux or different energy spectra has not been ascertained.

Following are observations made on physical properties of neutron-irradiated graphite.

Mechanical properties. Produces stronger, harder, more brittle material; more difficult to machine (*746*). Stress-strain curves become more linear; magnitude of residual strain remaining after application and removal of stress diminishes (*442*).

Thermal conductivity. Brief irradiations produce large decrease; continues to decrease less rapidly with prolonged irradiation; in some cases, conductivity reduced by factors greater than 50. Effects are markedly reduced at higher irradiation temperatures (*746*).

Electrical resistivity. Although not important in evaluating graphite as reactor component, it is an index of radiation damage. Changes rapidly during irradiation stages; rises quickly to maximum at about 2×10^{20} n/cm², decreases slightly, begins slow increase above 10×10^{20} n/cm²; effect reduced at higher temperature although less sensitive to temperature change than thermal conductivity (*746*). Transverse specimens show slightly smaller changes because of smaller effective crystal size (*442*).

Crystallite properties. Major effects are distortion of lattice and trend toward amorphous form; distortion involves expansion in C_0 (interplanar) dimension and shrinkage or warping of a_0 dimension along crystal planes. In X-ray diffraction studies, low exposures cause decrease in peak intensity and shift in diffraction angle; peak shape remains constant; above $4–6 \times 10^{20}$ n/cm² peak broadens and becomes very diffuse at high exposures. Highest uncorrected C_0 spacing observed is around 8.0 after 20×10^{20} n/cm² in MTR; effect is decreased markedly at higher exposure temperatures. Broadened X-ray diffraction peak is attributed to bending and slippage of crystal planes as crystallites break up; above 13×10^{20} n/cm² crystal structure similar to that of carbon blacks and decomposed organic solids (*746*). C_0 spacing increases almost linearly with dose at 1.3%/10^{20} n/cm² at 30° C (*442*).

Dimensional stability. Tends to expand along transverse planes and contract along parallel planes. (Transverse-cut sample made from natural flake graphite expanded 24% during 25×10^{20} n/cm²—an order of magnitude greater than for artificial graphites) (*746*).

Thermal expansion. Data indicates irradiation has negligible effect on temperature coefficient of thermal expansion of reactor-grade graphite (*746*). Coefficient does not increase by more than 5% for 10^{20} n/cm² at 30° (*442*).

Specific heat. Small increase when measured at temperatures below exposure temperature. Total stored energy (475 cal/gm) of one sample indicates exposure equivalent to 15×10^{20} n/cm² in cooled test hole; specific heat at room temperature was raised 0.006

cal/gm/° C (about 3%) by irradiation (*746*).

Stored energy. Irradiation increases heat content of crystal lattice and is observed as increased heat of combustion. Stored energy is released during annealing; all energy accumulated during brief irradiations is released by annealing at 1,000° C but essentially none accumulated after $>13 \times 10^{20}$ n/cm² is released by 1,000° C annealing. With release of energy, C_0 spacing decreases toward unirradiated value. At 200° C annealing, energy release peaks, which intensifies up to 4.2×10^{20} n/cm²; at higher exposures, peak diminishes; exposure at which shape of C_0 X-ray diffraction peak begins to broaden corresponds to exposure at which 200° C energy release decreases in magnitude. Irradiation at 111–164° C eliminates 200° C peak. The larger the crystallite size, the more the stored energy. Calorimetric methods are sensitive to experimental techniques and have been greatly improved; results of different workers must be compared cautiously (*746*).

Annealing physical-property damage. Quantitative recovery of changes can be attained as graphitization temperatures are approached. Irradiation damage previously incurred by irradiation at low temperature is annealed if temperature is increased during course of irradiation (nuclear annealing); nuclear annealing can effect recovery of damage at temperatures lower than those required by thermal annealing; effect due to dissipation of kinetic energy transferred from neutron of more than thermal energy (*746*). Similarity between recovery curves of Young's modulus and other properties suggests that changes are in crystalline properties rather than in bonds between crystals (*442*).

Water Decomposition

Water decomposition by radiation, due to the molecular part (H_2 and H_2O_2) of the primary decomposition yield, is reversed by a back reaction of free radicals with decomposition products (*738*). The back reaction is inhibited by impurities and an excess of H_2O_2 or O_2 in the decomposition products, but is favored by excess H_2. Thus the amount of decomposition depends on the ionization densities of the incident radiation.

The results of experimental work can be correlated qualitatively by the mechanism

$$H_2O = H_2,\ H_2O_2,\ H,\ OH$$
$$OH + H_2 = H_2O + H$$
$$H + H_2O_2 = H_2O + OH$$
$$OH + H_2O_2 = H_2O + HO_2$$
$$HO_2 + HO_2 = H_2O_2 + O_2$$
$$H + O_2 = HO_2$$

Decomposition by fission products. The rate of energy loss in a fission track is over an order of magnitude greater than for any other radiation. The initial yields of gas production for aqueous solutions of UO_2SO_4, UO_2F_2, $UO_2(NO_3)_2$, and $U(SO_4)_2$ have been determined as a function of various factors (*741*).

Where fissions contributed more than 99% of the total energy, the yield was independent of intensity. With constant uranium concentration and type of radiation (U^{235} not enriched), small differences accompanying changes in anion were traced apparently to differences in *p*H. For Co^{60} γ-rays an 18% decrease in H_2 yield has been found on going from *p*H 7 to 0.4.

The most striking effect on H_2 yield is the decrease with increased uranium concentration. This cannot be explained by *p*H changes; variation of *p*H from 2 to 0.1 for 100 gm/l uranium decreased the yield from 1.35 to 1.20, whereas for 800 gm/l uranium at *p*H 1.03, H_2 yield was 0.49. Adding comparable amounts of cesium sulfate had no marked effect, indicating that this decrease in H_2 yield cannot be attributed to energy absorption in the solute.

The mechanism by which uranium lowers the H_2 yield is not clearly understood. With cobalt gammas, uranium has little effect on radical formation. This conclusion is based on the observation that the H_2O_2 yield in water containing dissolved H_2 and O_2 during irradiation was decreased by uranium by the same amount as the H_2 yield was known to be decreased. If H_2 and H_2O_2 are only formed from H and O radicals in regions of high ionization density, the action of uranium must occur only in regions of high energy density (*741*).

NRX and D_2O experience. The reactor vessel holds 15,000 liters of D_2O with continuous flow of 25 l/min. (*7*). About 5 l/min pass through ion-exchange columns (strong-base anion and strong-acid cation resins) to remove ionic impurities. In another circuit, D_2O is pumped through heat exchangers at 900 l/min. With the reactor operating at 40 Mw the temperature of the D_2O is 50° C.

In 1948, at 10 Mw 2 gm D_2O decomposed per hour (2 liters of electrolytic gas per hour). In 1951, at 26 Mw, decomposition was 0.8 gm/hr. The drop in rate may be attributed to improved purity. D_2O_2 concentrations fluctuate considerably but are never greater than 5 ppm and are usually less than 1 ppm.

In the central irradiation thimble (thermal flux of 5×10^{13} n/cm²/sec), the stability of D_2O cooling experiments in closed loops has been studied. The total volume of D_2O is usually 200–300 liters at 600 l/min; 0.5–2 l/min is by-passed through ion-exchange columns.

The proper ion-exchange resin was not available for initial measurements of the closed-loop D_2O. At 60° C, Cl^- concentration was 5 ppm, D_2 evolution was 380 cm³ per liter D_2O per day, and O_2 evolution was half this. With proper resins, D_2 evolution fell to 6 cm³/l/day. After clean up of the D_2O, O_2 evolution fell to and remained at an unmeasurably low value of <0.1 cm³/l/day. D_2O_2 concentrations were high when gas evolution was high, but fell to <0.1 ppm with the proper resin.

At 70° C with dissolved gases removed from the loop by a helium stripper, D_2 concentration was 1 cm³/l and production rate 7 cm³/l/day. O_2 concentrations remained at 0.1 cm³/l or less and D_2O_2 below 0.2 ppm.

The results of these tests may be summarized as: If D_2O purity is maintained at a high level (specific conductivity of 1×10^{-6} ohm^{-1} cm^{-1}) decomposition is negligible. If D_2 is continously removed from solution, the D_2 formation rate is higher. O_2 and D_2O_2 concentrations are always extremely low. If D_2O is allowed to become impure, evolution of electrolytic gas at a high rate will be observed. If H_2 and O_2 are injected into D_2O under irradiation, extremely rapid recombination will take place.

Preliminary observations have been made from loop tests at up to 300° C and 130 atmospheres. At high temperatures, water is no less resistant to reactor radiation than at lower temperatures if it is pure. H_2 evolution rates are higher, dissolved O_2 concentrations are vanishingly small, and peroxides are unstable.

physics

Theoretical discussions of nuclear models
Improved neutron cross sections
Newly declassified data on the fission process
Counting and counters

There was a substantial amount of physics discussed at the Geneva conference even though that was not the primary subject. There were theoretical discussions of nuclear models and cross-section predictions from them. Much of the motivation has come from improved data on neutron cross sections (Fig. 1). Some of the theoretical work is reported here.

Weisskopf (*830*)* resurveyed neutron cross section calculations for a sharp-edge well. Much better agreement with measured cross sections is obtained if the nuclear radius is $(1.26\ A^{1/3} + 0.7) \times 10^{-13}$ cm instead of $1.45\ A^{1/3} \times 10^{-13}$ cm.

Nemirovski (*654*) spoke on the results of rounding the edge of the effective nuclear potential well. There may be some question regarding his approximations, but if one takes the results at face value, he has carried out important numerical calculation in the low-energy region (<10 Mev) which have so far not been done in the USA. He uses the wrong (20 Mev as opposed to 40 Mev) well depth however.

Bethe (*585*) discussed the relation between the fluctuation of neutron and fission widths from level to level and capture-to-fission ratios, α. He concludes that direct measurements of α and values of α inferred from an assumed distribution of widths are consistent.

Wheeler (*593*) reviewed many aspects of fission physics such as the variation from resonance to resonance of α. He suggests a probability distribution for $(1/\alpha)$ on the basis of the constancy of radiation width and fluctuation of fission width. Extrapolating stability criteria for heavy nuclei Wheeler concludes that nuclei *twice as heavy* as known nuclei can be created by massive neutron bombardment and will live long enough to be studied.

Nossov (*653*), using a model consisting of a prolate ellipsoid with an equatorial "squeeze," was able from stability considerations to obtain a fission-fragment mass ratio of 1.9. This is presumably the ratio of the most probable masses and is in good agreement with experiment considering the crude nature of the model. The point of view is quite different from and probably more realistic than that of Fong who obtained mass distribution curves without discussing intermediate distortional features.

A. Bohr (*911*) suggested that the thresholds for neutron fission of an even-odd nucleus may differ by as much as an Mev for different spins of the compound state and also that the ratio of fission to capture may be quite different for the two systems of levels.

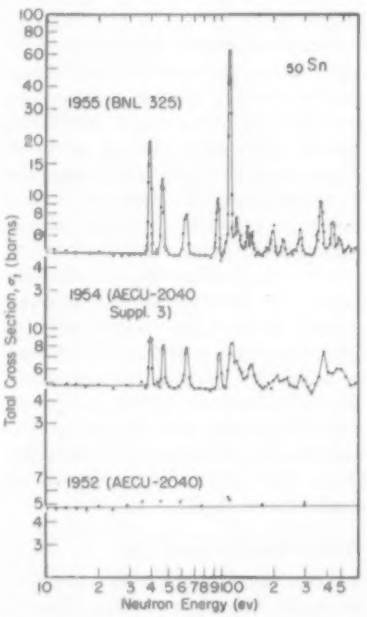


FIG. 1. Improvements in neutron-cross-section data over the years reflect improvements in resolution of velocity spectrometers and crystal monochromators. Further improvements of orders of magnitude is unlikely in these highly developed instruments; on the other hand, theory of level widths and distribution is sufficiently advanced to determine objectively if essentially all levels have been found. This curve is from Hughes' discussion (*576*) of neutron velocity selectors used at reactors, presented at the session on equipment and techniques used in measuring cross sections

* References are UN papers listed on p. 94.

Neutron Cross Sections

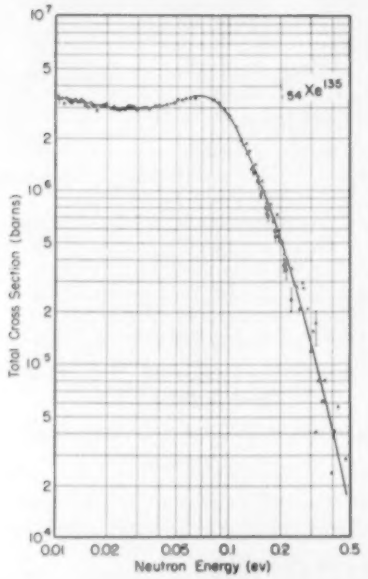


FIG. 1. Total cross section of Xe^{135}. Resonance fits these parameters: E_0 = 0.084 ev, $\sigma_o = 3.4 \times 10^6$ b, $\Gamma = 0.11$ ev

Reactor-materials cross sections. Hurwitz of KAPL (*590*) reviewed this field, pointing out that the accurate differential cross-section information now available (Fig. 1, p. 72) makes it possible to calculate some resonance integrals as accurately as they can be measured. Integral measurements remain vital, however, especially for strong-self-shielding cases where cross-section data at energies above those where accurate data are now available may be required.

Typical are the following improved thermal cross sections:

	Cross section	
Isotope	*Type*	*Value (barns)*
B	Total	755 ± 4
C	Capture	0.0032
Zr	"	0.18
111-dTa^{182}	"	47,000
Pa^{233}*	"	66 ± 7

* Pa^{233} is important because it is product of neutron capture in Th^{232} and one doesn't want it to capture another neutron and become Pa^{234} (which decays to unfissionable U^{234}) if it is to decay to fissionable U^{233} in Th breeder reactors.

Xe^{135} total cross section was measured by Bernstein and Smith, ORNL (*591*). Measurements were made in 1954 with a mechanical velocity selector. Two samples of about 500 curies (2×10^{18} atoms) each were drawn from off-gases of the Homogeneous Reactor Experiment and transmission measurements run. About 1,500 transmissions were measured on the 9.2-hr half-life sample. The mass of data was analyzed with the ORNL computer and gave the curve shown in Fig. 1.

Pu^{240}, which has a resonance at E_0 = 1.06 ± 0.02 ev with $\sigma_0 = 1.7 \times 10^5$ barns and $\Gamma = 0.045 \pm 0.015$ ev, is significant because it is produced in considerable quantity in thermal reactors containing Pu^{239}.

Fundamental Thermal-Neutron Cross Sections

Very many slow-neutron measurements are made relative to the standard cross section of another element. J. A. Harvey of Brookhaven discussed these basic cross sections (832).

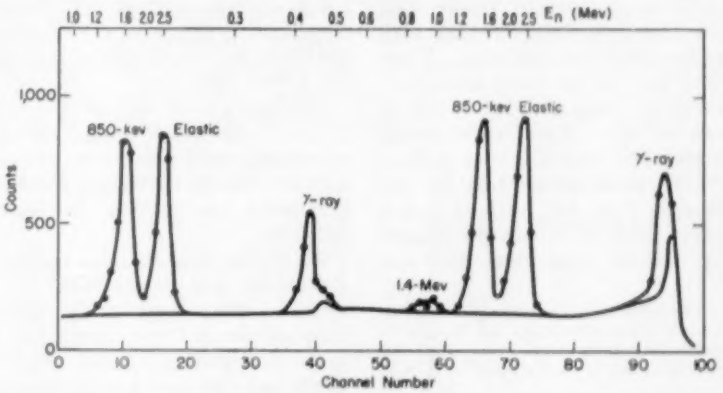
A good standard should: (a) have an absorption cross section much larger than its scattering cross section, (b) have an absorption cross section that varies inversely as the velocity of the neutron in the thermal-energy range, (c) be monoisotopic, and (d) be readily available and easily handled.

Here are the present best values for the well-known standards, boron and gold. *See p. 74 for data on reactor-significant U^{235} and U^{238}.*

Boron *satisfies all requirements except (c). There are 2% variations in the relative abundance of its isotopes. Hence for precision work it is necessary to stockpile a standard boron sample. The present best U. S. value for the absorption cross section is* **752 ± 3** *barns at 2,200 m/sec for B that has 0.185 ± ½% B^{10}. Unfortunately British measurements of the same B gave 771 ± 6 and Swedish measurements gave 764 ± 3; thus there is a need for further intercomparison.*

Gold *is a classic standard, suffering only from a slight departure from 1/v dependence. Measurements at Brookhaven give* **98.7 ± 0.6** *barns, in good agreement with Egelstaff's Harwell value of 98.4 ± 0.9* [J. Nuclear Energy **1**, *57 and 92 (1955)*]. *The f-factor is 1.005; thus when gold is used as a standard for reactor oscillator measurements a value ½% larger than the preceding 2,200 m/sec value should be used.*

Inelastic Neutron Scattering from Fe at 90 Deg



Note well-known strongly excited 850-kev level in Fe^{56} as well as weaker 1.4-Mev level in Fe^{54}. This typifies Cranberg's work applying time-of-flight techniques to fast-neutron measurements using fast organic scintillators and millimicrosecond electronics (*577*). Over-all cross-section accuracy is ~5%; minimum detectable cross section for monoisotopic element is ~4 milliborns/steradian

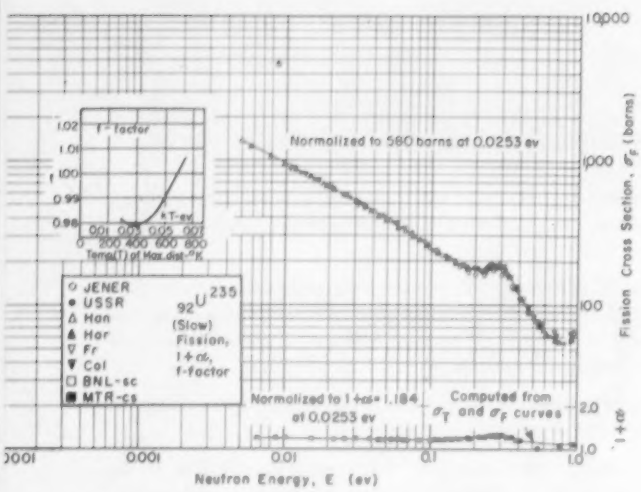


FIG. 1. Slow-neutron fission cross section of U^{235}

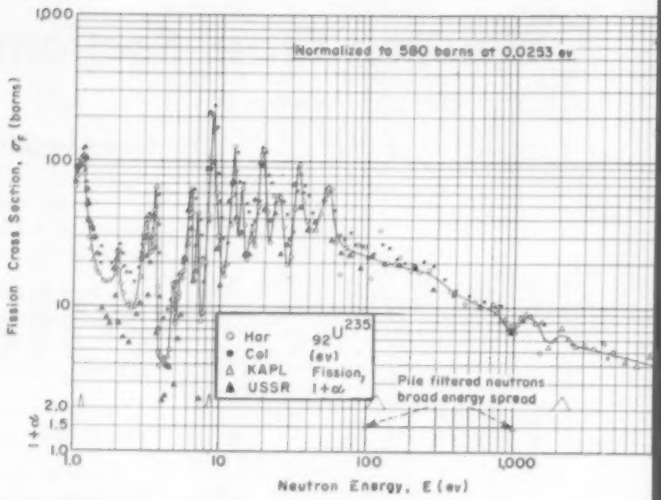


FIG. 2. Fission cross section of U^{235} in the resonance region

Data on Fissionable Elements

The Geneva conference has produced a flood of hitherto secret data on the fission process. High-resolution crystal- and velocity-spectrometer cross-section data for scattering, capture, and fission in U^{233}, U^{235}, U^{238}, Pu^{239}, etc., were presented by France (*354, 355*), USA (*589, 586*), UK (*422, 423*), and the USSR (*645, 646, 641*). There is also data on capture-to-fisson ratio, number of neutrons per fission, delayed neutrons, etc. (Late world-average cross sections are in ROUNDUP, p. 9.)

Fission cross-section data are presented in Figs. 1–4, adapted from a report* prepared by the Brookhaven cross sections group from data submitted by the USA and other governments in advance of Geneva. The only changes made at Brookhaven concern the normalization of fission cross sections to reasonable world "best values" at thermal. The USSR and UK fission cross sections of U^{235} were normalized to 580 barns at 0.0253 ev, the USSR Pu^{239} to 720 barns, and the UK U^{233} to 533 barns.

The fission cross sections are now known in considerable detail. Several papers analyzed the resonances according to the Breit-Wigner formula. Parameters are available for many resonances.

U^{233} fission, capture cross sections. Kukavadse, *et al.*, of the USSR (*644*) measured the absorption and fission cross sections of U^{233} by irradiating a sandwich consisting of a 60.75-mg foil of U^{233} and a 28-mg foil of Li in a reactor. Neutron flux is monitored by the decrease in Li^6 concentration due to $Li^6(n,\alpha)H^3$ while radiative captures, $U^{233}(n,\gamma)U^{234}$, are determined by mass spectrographic analysis for U^{234}. Total decrease in U^{233} is determined by decrease in α activity and by mass spectrometry.

Based on a Li^6 capture cross section of 930 barns, and accounting for self-shielding, the capture cross section of U^{233} is 615 ± 30 barns.

Number of neutrons/neutron absorbed, η, has been measured by a new method, reported H. Palevsky of Brookhaven (*587*). One counts, as a function of energy, fission neutrons from a thick sample placed in a neutron beam whose flux is measured with a thin BF_3 detector. Then η is proportional to $C_U/\sqrt{E_i}\,C_B$ where C_U and C_B are the counting rates of the fission-neutron scintillation detector and flux-measuring BF_3 detector respectively. Data are then normalized at 0.0253 ev to other absolute measurements.

Power of new method is that it measures η directly; this permitted first direct determination of the fact that η increases above 0.29-ev U^{235} resonance. Opposite result was anticipated on basis of earlier measurements of σ_c/σ_f at thermal energies and at 0.29-ev resonance.

Normalized data for U^{235} (Fig. 5) agree within 1% with that calculated

* The report is an addendum to BNL-325, Neutron Cross Sections, by Donald J. Hughes and John A. Harvey. BNL-325 constitutes the second edition of AECU-2040.

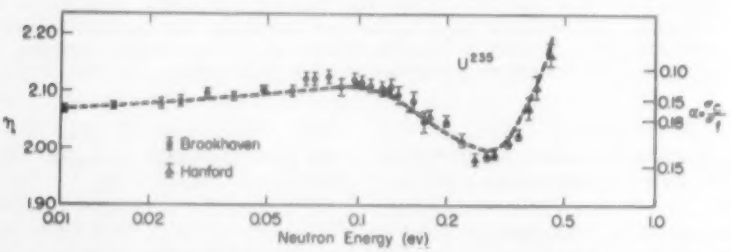


FIG. 5. Energy dependence of η (number of neutrons per neutron absorbed) for U^{235} normalized to $\eta = 2.08$ at 0.0253 ev. Dotted line is calculated from ratios of fission to absorption cross sections given in BNL-325*. Bottom α ordinate should be 0.25

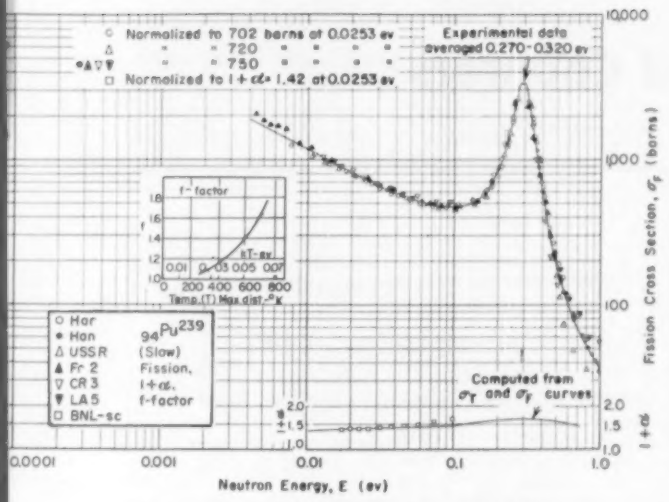


FIG. 3. Slow-neutron fission cross section of Pu^{239}

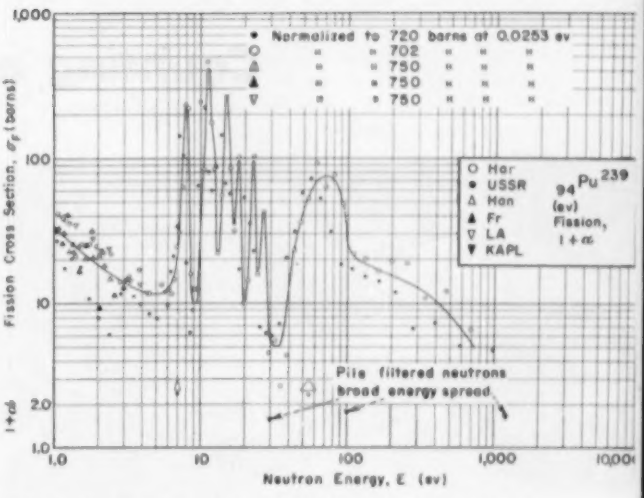


FIG. 4. Fission cross section of Pu^{239} in the resonance region

from ratio of capture to fission cross sections. Some disagreement persists in case of Pu^{239}. Agreement in case of U^{233} is better than 5% statistical errors in fission cross section merit.

Capture to fission ratio of U^{235} and Pu^{239} for intermediate-energy neutrons, as reported by Kanne, Stewart, and White of KAPL (*595*), were measured in 1947, using mass spectrometric methods. After pile irradiation, capture-products U^{236} and Pu^{240} could be determined from spontaneous-fission counts and mass spectrometric analysis respectively, giving total number of captures. The total number of fissions was determined from quantitative analysis of fission-product yields.

One purpose of Pu^{239} measurement was to determine if high α for thermal neutrons and at 0.3-ev resonance might not decrease at intermediate energies and thus make Pu breeder feasible . . . but the strong capture by Pu^{239} revealed here precludes a practical intermediate-energy breeder reactor.

Delayed-neutron groups from all fissionable elements seem to have the same periods, suggesting that the same delayed-neutron precursors are produced in fission regardless of the fissioning element reported Los Alamos' Keepin (*831*). (Further data will appear in an article by Paxton scheduled for the October '55 NUCLEONICS.)

Number of prompt neutrons/fission. Using a large liquid-scintillation 4π neutron counter with 82% efficiency for detecting fission neutrons, Los Alamos researchers (*592*) measured the number distribution of prompt neutrons emitted in U^{235} fission (Fig. 6): average number is 2.46 ± 0.03 at thermal energies. For higher energies one has $\bar{\nu}(0.7\ \text{Mev})/\bar{\nu}(\text{th}) = 1.02 \pm 0.02$ and 1.15 ± 0.14 for $E_n = 1$ Mev.

Prompt gamma rays are emitted by primary fission products after all energetically allowed prompt neutrons have been emitted. Gamble and Francis at Oak Ridge (*592*) have observed the spectrum of gamma rays in coincidence with fission using a NaI scintillation spectrometer (Fig. 7). Since they also count gammas from the inelastic scattering of prompt neutrons, the numbers and energies of gammas are overestimated. The average energy of prompt gamma rays from fission is expected to be less than the 7.5 Mev obtained from Fig. 7. Similarly, the average number of prompt gamma rays per fission should be less than the 7.5 obtained from Fig. 7.

Energies of prompt neutrons from fission has been measured with a cloud chamber below 700 kev, by time-of-flight techniques in the 0.2-3-Mev range, and by photographic-plate recoil spectrometry in the 0.3-12-Mev region. These Los Alamos measurements (*592*) have better statistics than earlier work. They fit the semi-empirical expressions $N(E) \sim (\sinh \sqrt{2.29E}) \exp(-E/0.965)$ and the simpler $N(E) \sim \sqrt{E} \exp(-E/1.29)$.

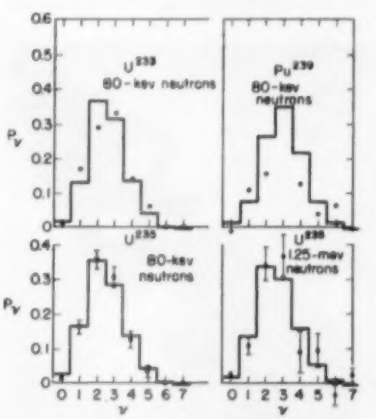


FIG. 6. Calculated and observed variations in ν for neutron-induced fission at energies shown. Statistical uncertainties in U^{233} and Pu^{239} data considerably exceed those for U^{235}. Theory is based on mass equation of fission, neutron evaporation considerations, and mass surface of nuclides

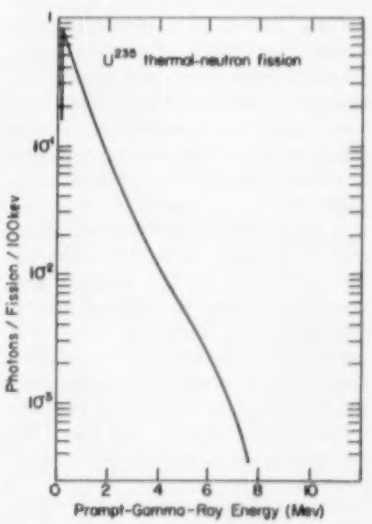


FIG. 7. Prompt gamma rays from fission as measured by Gamble and Francis at Oak Ridge with scintillation spectrometer

Counting and Counters

FIG. 1. Human counter is example of liquid scintillator used as 4π counter. Central hole is 46 cm in diameter; 108 photomultipliers are used to give maximum optical efficiency for bremsstrahlung detecting

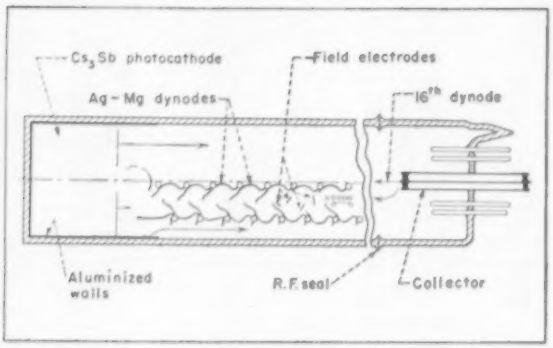


FIG. 2. H6687 developmental photomultiplier. Field-forming electrodes are operated at more positive potential than dynodes with which they are associated to increase space-charge-limited current to >1 amp

In several sessions of the Geneva Conference advances in nuclear instrumentation and techniques were discussed. The instruments ranged in size from the Los Alamos human counter, Fig. 1, to miniature germanium probes for fast-neutron dosimetry (*60* and NU, March '55, p. 58). Here are some of the highlights.

Liquid scintillation counting of natural radiocarbon. F. N. Hayes, *et al.* (*63*) submitted a paper on the greater sensitivity and range afforded by the high counting efficiency and large sample size possible in liquid-scintillation radiocarbon assay. Sample-preparation difficulties have held up application of this method. Ideally one would like to synthesize the solvent; then a major portion of the solution is derived from the sample and no loss in pulse height is incurred due to the introduction of quenching substances. However the difficulty in total synthesis of, say, toluene from CO_2 is such that researchers have resorted to partial synthesis. For example, toluene is made from CO_2 and inactive phenylmagnesium bromide. In this case, only the methyl group is derived from the carbon of the sample, and hence the counting solution is 13% (by weight) sample.

Sometimes one synthesizes a "diluent," a substance compatible with the solvent with a minimum of light quenching or absorption. Arnold has combined an acetylene synthesis used by Suess with a synthesis discovered by Nieuwland leading to a mixture of hexane and octane via copper acetylide. The mixture of aliphatic hydrocarbons obtained can be added to the scintillation system in concentrations as high as 60%, which is equivalent to about 50% by weight sample carbon. The synthesis is complex, but is now being used routinely. Yields are about 30%, but it has been proved by careful C^{13} measurements that isotopic fractionation in the synthesis is very slight.

FIG. 3. "Phoswich" detector is slotted plastic scintillator (p-terphenyl in polystyrene) whose machined slots can be filled with element whose threshold reaction provides neutron sensitivity

Hayes by-passes the difficulty of synthesis by suitable selection of the sample. An excellent scintillation solvent, *p*-cymene (*p*-isopropyltoluene), can be easily prepared from a large number of natural terpenes. The terpenes are widely distributed in essential oils found in the wood, leaves, and fruits of trees, shrubs, and even grasses. "Dead" cymene for background measurement is prepared from petrochemical propylene and toluene.

Recent developments in photomultipliers. G. A. Morton of RCA Laboratories (*61*) unveiled some developmental photomultipliers in a review of scintillation counting. The high-output 14-dynode H6687, akin to the developmental C7187A, is shown in Fig. 2. It has a 14.2-cm^2 photocathode with S-11 response. Its field-forming electrodes permit output currents >1 amp, making it possible to operate coincidence circuitry etc. without amplifiers.

Larger photomultipliers are represented by the 9-in. H6699, which uses a 10-stage linear multiplier structure.

Interesting new photoemissive materials based on Na and K systems were discussed. Besides representing new highs in sensitivity (to 200 μa/lumen) some of the new materials avoid the high vapor-pressure and low-work-function difficulties often associated with cesium-bearing photocathodes. More remains to be done here; in par-

ticular the strong infrared response makes cooling necessary.

Fast-neutron dosimetry. Threshold detectors have been used to monitor neutron doses of 10^9 n/cm² delivered in a short interval, reported G. S. Hurst (*65*). The following elements, which have cross sections that rise rapidly at threshold and remain essentially constant thereafter, were used:

Detector	*Threshold energy (kev)*
Pu^{239} shielded with B^{10}	0.1
Np^{237}	700
U^{238}	1,500
S^{32}	2,500

Using these together with gold for the thermal region one obtains a reasonably detailed histogrammic representation of the neutron spectrum. This method has been used at weapons tests and in reactor irradiation of test animals.

Cowan and O'Brien of Brookhaven have applied high-sensitivity threshold detectors to health-physics work (*63*). Essentially they elaborated on schemes incorporating neutron-detecting elements in or near a scintillator. Thus they can count fission or induced activity with great sensitivity. Figure 3 is a photograph of a typical detector.

Currently they are investigating Kallmann and Furst's Bi-bearing liquid-scintillator system: 3 gm of α-napthylphenyloxazozole per liter of solvent consisting of 70% xylene, 24% napthalene, and 6% triphenylbismuthene by weight. A 10-l tank will contain 370 gm of Bi and have about 50% scintillation detecting efficiency; this will give a counting rate of about 20 cpm/n/cm² as contrasted with 0.04 cpm/n/cm² for large Bi fission chambers.

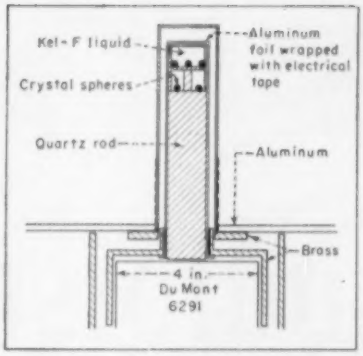


FIG. 4. Plastic-sphere scintillation detector of fast neutrons. Sphere diameters are such that most recoil protons dissipate their energy within them, while recoil electrons from γ-ray processes have range much in excess of sphere diameter. Nonhydrogenous Kel-F separating spheres keeps electrons from traversing more than one sphere

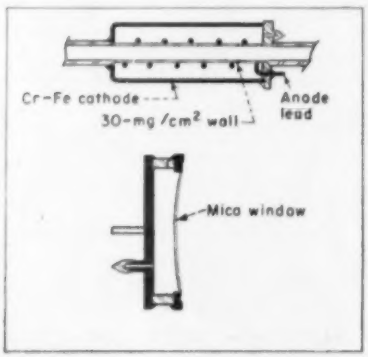
New advances in neutron detection were reviewed by Bollinger (*580*). For slow-neutron detection with time-of-flight spectrometers the following are used: (a) Boron-loaded liquid scintillators (4 gm/l of phenylbiphenyloxadiazole and 20 mg/l of POPOP in equal parts of toluene and enriched methyl borate); mean neutron capture time is 0.6 μsec. (b) Arrays of BF_3 counters 2.2 cm in diameter and 23 cm long within a common pressure vessel. These give 0.4-μsec collection time, but counter length gives some uncertainty as to time of arrival. (c) Intimate mixtures of 2 gm of enriched B_2O_3 and 1 gm of ZnS, fused and formed into translucent scintillating discs of 20-cm² area, as developed by Palevsky at Brookhaven. Neutron capture time and pulse rise time are both less than 0.1 μsec. Thicker discs cannot be used because of low optical transmission.

For fast neutrons there is Rice Institute's ingenious scintillation counter, illustrated in Fig. 4, that discriminates well against gammas. Furthermore, the counter is neutron-energy sensitive, a valuable property for measurements of inelastic neutron scattering —for appropriate sphere size and accepting only pulses that are >85% of the maximum obtained with 3.5-Mev neutrons, relative counting efficiencies are: 100% at 3.5 Mev; 29% at 3.0; and 3% at 2.7.

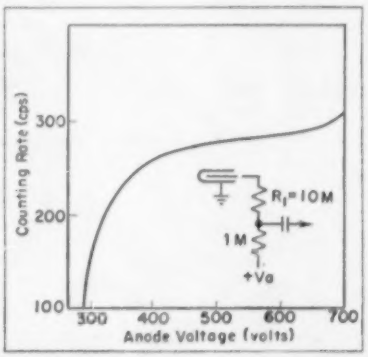
Also discussed were the giant cadmium-loaded liquid-scintillation counters developed by Cowan and Reines at Los Alamos and a complex system of liquid-scintillation counters used with the fast chopper at Argonne.

New Forms of Geiger-Mueller Counters

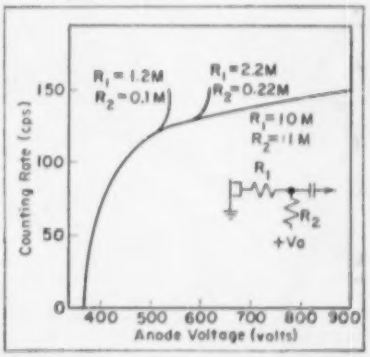
Geiger counters with large-diameter anodes provide the first substantial change in G-M tube geometry since 1908, report Hermsen et al. *of Philips Research Labs synchrocyclotron group (930). Some new designs are shown here.*



LIQUID-FLOW beta counter (top) measures activity of liquid flowing in thin-walled pipe-anode. Parallel-plate Geiger counter is shown at bottom



CHARACTERISTIC of liquid-flow counter. Dead time decreases as voltage increases and is ~200 μsec. Low-voltage halogen filling gases are used



PARALLEL-PLATE counter characteristic. Plateau slope is 5%/100 volts for R_1 = 10-Megohm. Dead time increases with increasing R_1

REPORT from GENEVA

radioisotopes

Material tracing and leak detection—wear tests—improvements in beta, gamma, and neutron gages—uses in therapy and diagnosis

Examples of Isotope Uses

Radioisotopes are the most immediately realizable benefit from atomic energy developments. Familarity with radioactivity for a pool of professional people is a benefit added to the *per se* advantages of isotope utilization. In USA industry alone, the total savings for all isotope applications are estimated at $100 million annually, and are expected to grow 10-fold in 10 yr. Speed in gaining new knowledge is an additional advantage.

Power of radioisotope tracers. Limit of detection for H^3, a most favorable stable tracer, is 10^6 dilution, obtainable with difficulty using a mass spectrometer. Using latest low-level counting techniques, carrier-free P^{32} can be detected after 10^{17} dilution with ordinary P. Theoretically, C^{14} can be traced after 10^{12} dilution in "living" C. Sensitivity for H^3 is even greater. Isotope labeling also is unquestionably specific. Although there is an isotope effect on reaction rate, such effects can be allowed for in quantitative work, if necessary (*308*).*

TABLE 1—Typical β-Sources for Industrial Uses (*461*)

*Isotope**	*Source construction†*	*Dimensions (cm)* length × width	*Screen* (mg/cm^2)	*Strength* (*mc*)	*Strength* (mc/cm^2)
Pm^{147}	Ag strip sand., Pt face	5–10 × 0.3–1.25	5	5–20	0.5–1
Tl^{204}	Electrodeposit on Cu; Cu, Cd overdeposit	2–30 × 0.5–2.5	15	80 max.	1 max.
Sr^{90}‡	Ag strip sandwich	2.5–30 × 0.3–2.5	30–50	500 max.	250 max.
Ce^{144}	Ag strip sandwich	2.5–30 × 0.3–2.5	50	5–20	1–6
Ru^{106}¶	Same as Ce^{144}				

* Listed in order of energy. Usable for absorption thickness gaging up to these thicknesses: Pm-10, Tl-120, Sr-500, Ce- and Ru-700–1,000 mg/cm^2.

† Finished sources are tested for: strength-ionization chamber; uniformity-autoradiography with process film or contact printing paper, or fluorescent screen; external contamination-swab faces with solvent, removable activity <0.01 μc; leachability-immerse for 8 hr in 50° C distilled water, water activity <0.05 μc.

‡ Energetic β-emission is by Y^{90} daughter. Sr^{90} is made into medical applicators using circular or square plaque mounting of Ag-sandwich element. Surface dosage is 100–350 rep/min. Sr^{90} is made into eye applicators, using Ag foil mounted in polyethylene, with dosage to 180 rep/min.

¶ Ru^{106} is used for medical purposes in source same as used for Sr^{90}. Surface dosage is 180–600 rep/min.

Tracing by Radioisotopes

Salts and dyes were often used in flow tests before isotopes were available. However, density currents caused by brine solutions and dye adsorption on surfaces, or possible reactions to cause color changes, make radioisotopes superior. Substantially all the radiotracer can be accounted for, which is not the case with dyes and salts. Ru^{86}, Sc^{46}, and I^{131} have been found most useful in flow testing. Sr^{89}, except for its low tolerance concentration, would be a good tracer.

Floc penetration in sand filters. I^{131} was used to determine depth of floc penetration into rapid sand filters. Mutual coagulation between colloidal NaI and hydrous Fe_2O_3 allowed making a γ-ray profile with a G-M tube. Effect of rate of filtration and sand size was well studied for first time (*226*).

* References to UN papers on p. 94.

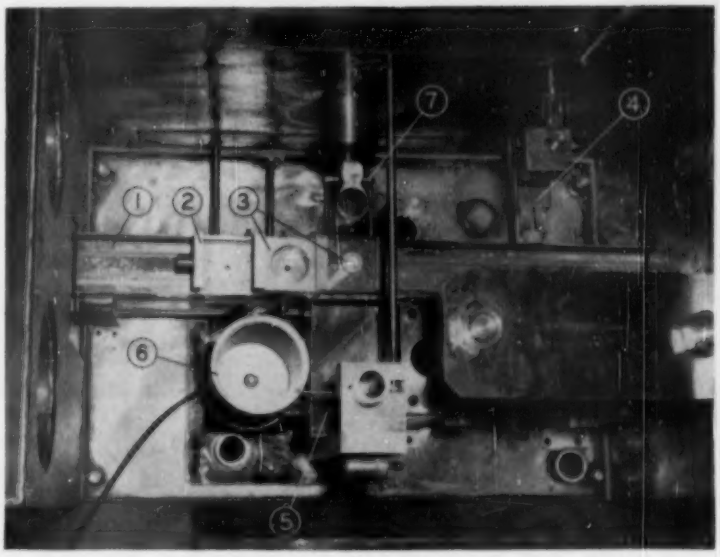
Mohawk River tracer test. By dumping 4.5 c of P^{32} into the KAPL outfall during a 3-hr period, and collecting samples at stations up to 15 miles downstream, the amount of low-level waste that KAPL could discharge was determined. One-liter samples were collected, evaporated to dryness (70 mg residue), and counted in end-window G-M tubes. About 5 days were needed for the peak activity to travel 7.5 miles. Peak was 450 dpm/l (*226*).

Leak detection in water pipes. In three methods, Na^{24} is used to find leaks in buried pipes up to 5 km long. High tolerance level (8 mc/l) and 15-hr half-life add to safety of tests. Methods are: (*a*) Fill pipe uniformly with radioisotope solution, close all outlets and raise pressure for ½ hr, flush out solution, then probe for activity near pipe. Smallest leak found this way was 2 ml/hr; routine detection spots 100 ml/hr leaks. (*b*) Longer pipes with higher leak rates are tested by closing all outlets, introducing high specific activity solution, and timing tracer flow by counters placed at convenient points, such as hydrants. At leak location there is an abrupt change in velocity. (*c*) Small leaks in long pipes are located by closing the valves and injecting ½ mc at pipe midpoint. Most tracer flows towards the leak, and counters on each side indicate the direction. Repeated application localizes the leak.

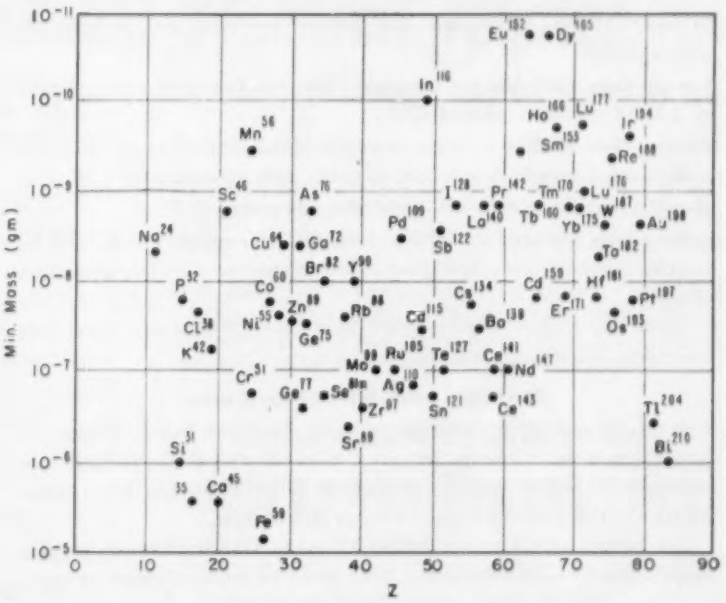
For long pipes this method has been successful. Tracer solution is fed to the pipe, followed by wash liquid, followed by a G-M detector with output to a small wire recorder, both carried on a "go-devil." Outside the pipe a series of 0.2 mc Co^{60} sources form a distance marker record. When the wire recorder is removed it is played back into a recording ratemeter; leaks are easily seen. Length effectively tested is 20 km; it will become much longer (*462*).

Radiometric titration techniques. Drinking-water analysis for chlorides uses labeled $AgNO_3$ and counting of solution. After stoichiometric amounts are added, activity increase is linear. Reverse-type method works for sulfates. Eliminated is the inaccuracy of visual-end-point methods, or 24-hr delay of gravimetric sulfate (*233*).

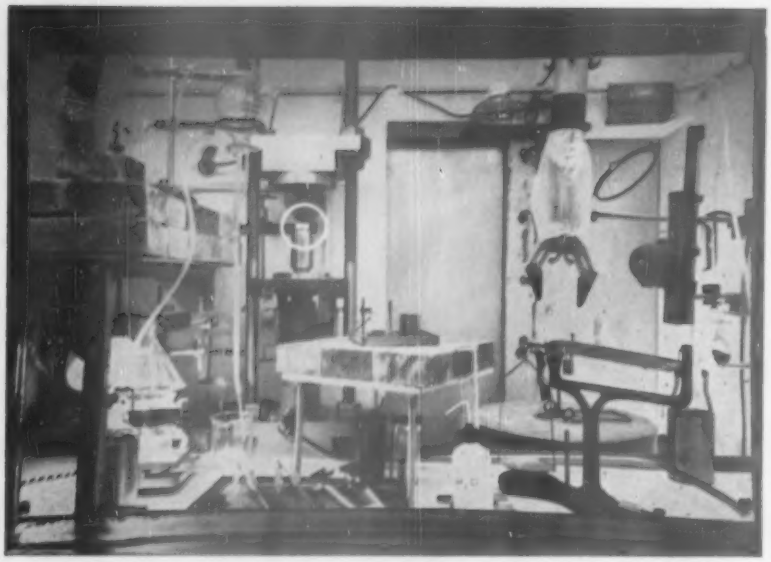
Low-grade ore concentrates in blast furnaces. The question of how much fine-powder ore would remain in



TOP VIEW of stationary hot cell for safe handling of Co^{60} pellets shows slide (1), transfer tray (2), inserts for transfer tray (3), tongs (4) for transfers from slide draw to insert, pellet reservoir (5), vibrating spiral conveyor (6) for loading sources, and loading tongs (7). Press (top shown on right) can either seam-weld loaded capsules or shear off capsule lids for unloading. Shielding is 30 cm of lead (13)



NEUTRON-ACTIVATION sensitivity depends on activation cross section of element, neutron flux, time of irradiation, and radionuclide half-life. Illustration shows minimum quantities determined with 5×10^{11} slow neutrons/cm²/sec for one month exposure in case of long-lived isotopes and for saturation time in case of short-lived (not less than 25 min) isotopes, i.e., corresponding to 40 dps (632)



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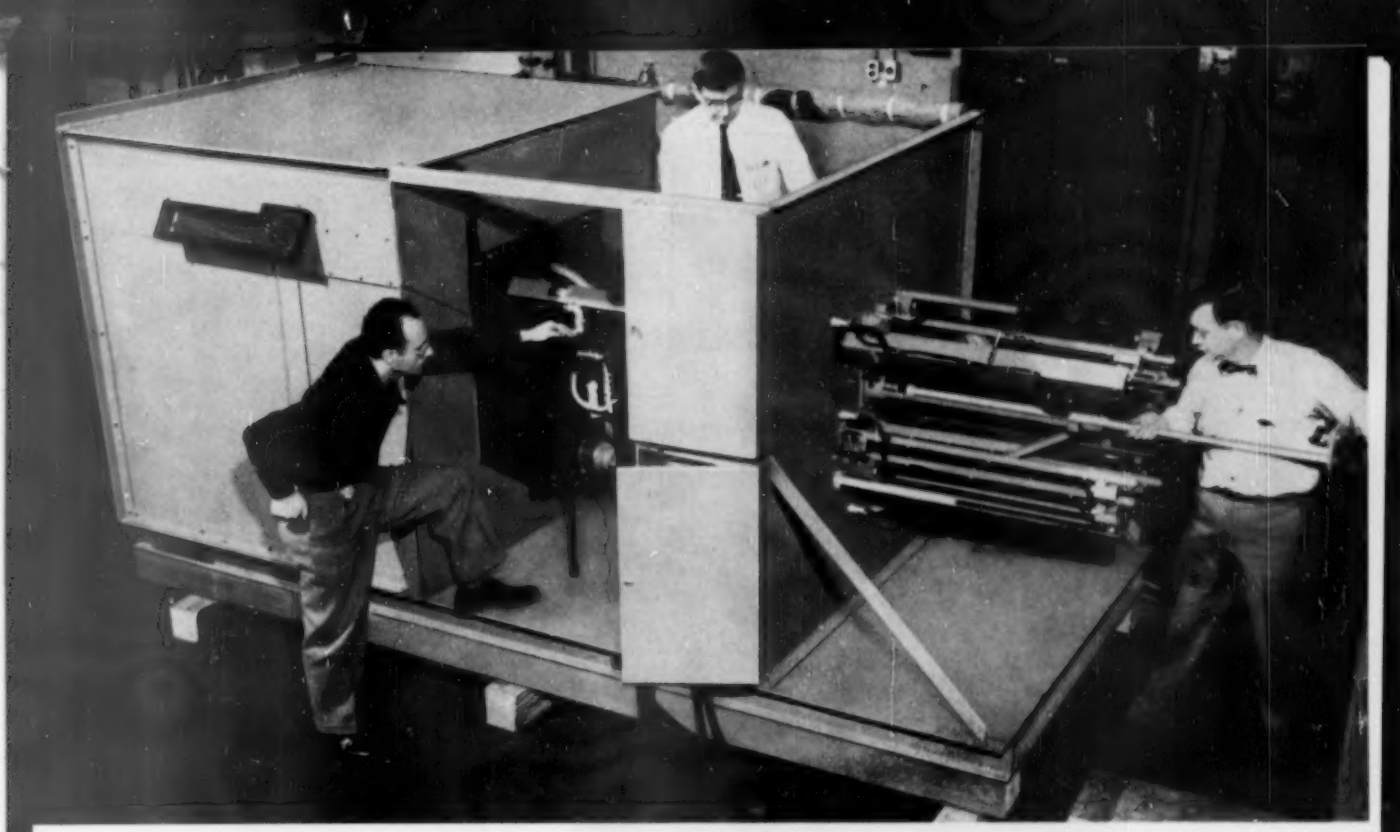
a blast furnace was solved by mixing 5 lb of finely powdered irradiated ore with 22 tons of nonirradiated powder and 54 tons of a different and coarser type. Samples of pig iron, slag, and expelled dust were tested. Surprisingly, 60% of the tagged ore remained as pig iron, but this was not high enough to be commercially practical. Further research was proven necessary in the field of iron-particle agglomeration so that low-grade concentrates can be used in blast furnaces (*146*).

Auto tires measured for wear. Tagging the treads of test tires with a radioactive plasticizer (triphenyl phosphate synthesized from P^{32}) and then measuring the radioactivity of the rubber adhering to the road surface has demonstrated that life expectancy at 60 mph is only 57% of that at 30 mph. Starting and stopping causes even greater wear. Formerly, tires had to be run at least 5,000 miles before wear became measurable, and determining stop-and-go loss was barely possible (*146*).

Iodine addition to salt cake. To test best way of adding I to salt cake to prevent leaching in rain, yet allow the animal to absorb it, various chemical forms of iodine were prepared from I^{131}. Measuring the radioactivity in rainwater and in the animal's vitals and excreta showed dithymol diiodide to be the best (*146*).

Sulfur content in coal. To find whether pyritic or organic S in coal was more important in determining total S content of coke, pyritic sulfur was synthesized with S^{32} and mixed with 10 tons of powdered coal containing ordinary pyritic and organic sulfur. For the first 6 hours, pyritic sulfur evolved more slowly than organic but faster for the next 7 hours. The net result was an almost equal rate of evolution. Thus, under normal coke-oven conditions, there is no advantage in using coal with a high ratio of pyritic to organic sulfur to produce low S coke (*146*).

Uses in steel making. Russian use of radioisotopes in ferrous metallurgy has been extensive. Theoretical topics studied include kinetics of desulfurization, and dephosphorization; Ca and Na transfer in slag melts; mechanism of admixture separation during steel solidification; and vapor pressure of Fe-P alloy. Studies on plant equipment include movement of charge and gases in blast furnaces, and melting of alloys added to open-hearth furnaces. Radioisotopes are



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used in blast-furnace-lining wear metering, determining P_2O_5 content of slag, measuring sheet thickness, and directing strips through operations according to grade, by tagging each strip differently (*707*).

Wear testing of three parts simultaneously. To define wear in two or three parts simultaneously, radioisotopes of different energy, or alternately, β- or γ-emitters may be applied. By either filtering out the β-rays or using various absorbers, counting of each isotope may be done. Wear for each part results. Radioisotope may be added by electroplating, casting the part with tracer, placing radioactive inserts in the part, or reactor irradiation (*713*).

Tracing chemical reactions. Course of chemical reactions, and whether homolytic or heterolytic reactions take place has been determined using C^{14}, H^2, S^{35}, and stable N^{15} and O^{18} as tracers. Technique is to label one of the reactants, and compare specific activity vs time for reactant, intermediates and final products. Oxidation of ethylene, butane, and methane; and rubber vulcanization are examples of reactions studied (*708*).

Gaging by Radioisotopes

In gaging applications, materials affect the radiation transmitted or reflected, providing an instrument signal. Because β-, γ-, or neutron radiation can be used, there is much versatility.

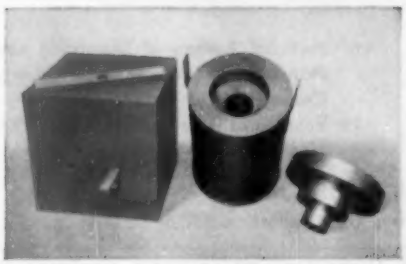
Beta-ray sources. Beta transmission gages are used routinely for measuring thickness of materials 1–1,200 mg/cm² thick. For best accuracy, half the β-rays should be absorbed. Material thicknesses from $0.2–4 \times t_{½}$ for a given source can be measured, with 1% accuracy obtained at $0.5–2 \times t_{½}$ (*463*).

H/C ratio gages for hydrocarbons are in use. Basis is that electron density of H is twice that of C.

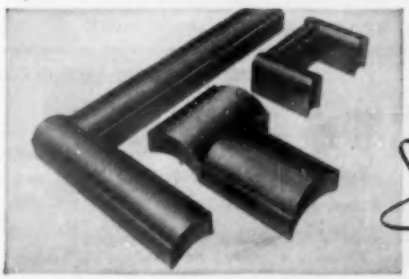
C^{14} is used to measure thickness of infrared spectrophotometry samples by placing sample between an electroscope and a 1.5-cm-diameter $BaCO_3$ source. Linear relationship exists for *ln* discharge time vs thickness for Cellophane and Al foil.

Automatic machine control using output of β-thickness gage as primary signal resulted in reduced product variation: Fourdrinier paper machine, 12% (before) to 3%; tandem cold-roll metal mill, 10% to 1.5%; rubber sheeting calender, 10% to 3.2%; rubber

For high density in small volume... For maximum adaptability... Lead



Typical lead shielding items – lead safe (cube), lead cask and extruded curved-joint lead brick.



Lead shielding has greater density (11.35 at 20°C) than any material anywhere near its cost—a worthwhile advantage when *space* and *money* are limited.

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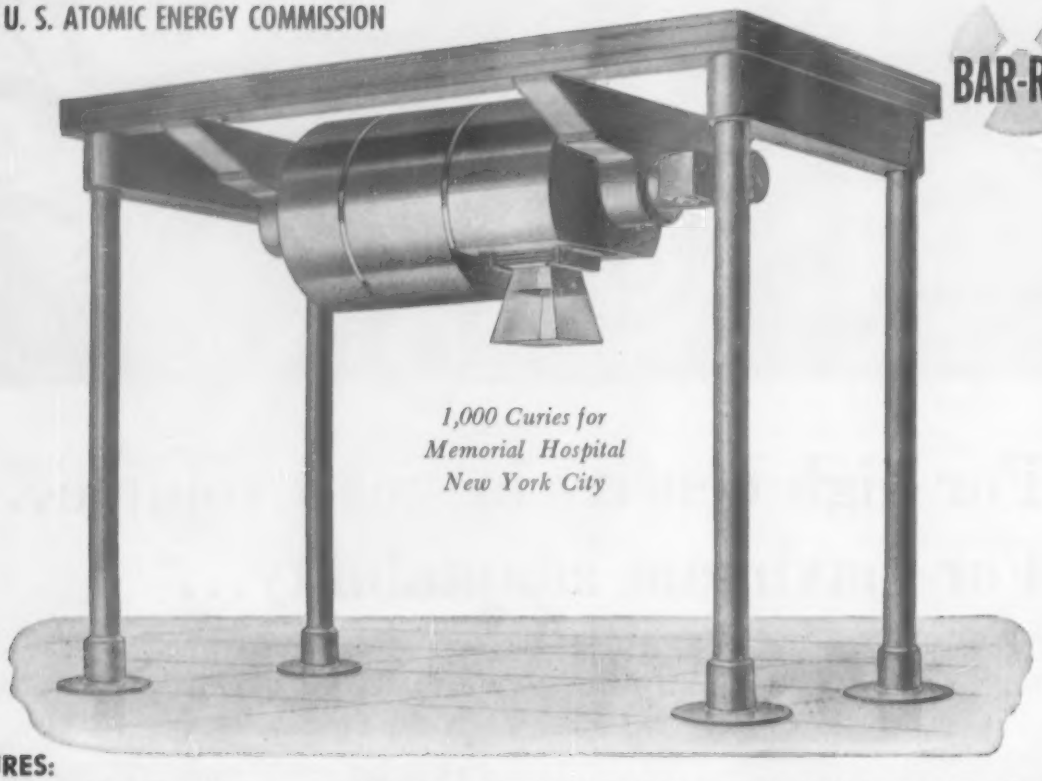

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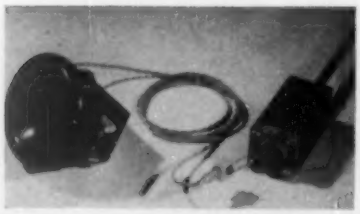

SEMICONDUCTOR NUCLEAR BATTERY (right) powering experimental transistorized audio oscillator. Single silicon junction, 0.5-cm diameter, irradiated by 50 mc of Sr^{90} provides 10 μa at ½ volt. Until isotopes with low cost, low energy, and long half-life are available, radioactive batteries will be limited to such applications (*169, 171*)

tire-fabric calender, 6.5% to 3.5%; plastics calender, 7% to 2.5%; saturator dip coater, factor of 2.5 improvement; abrasive making, factor of 2 improvement.

Backscatter is used to measure Zn-coating thickness on 15-mil-thick steel. Accuracy of monitoring is 0.06 oz, obtained by holding measuring head to strip separation constant within 60 mils in the ½-in. separation. Solution concentration measurements, using backscatter caused by Cu, Ag, or Pb ions, show promise (*164*).

Gamma-ray measurements. Petroleum fuels of only 0.3% density difference can be distinguished, using γ-transmission gages. A simplified form of this gage can be used as a level detector, and coupled to an automatic reject (*463*). One variation of density monitoring is used to control successive phase separations with density differences of 0.28, 0.15, and 0.07 (*164*).

Other materials are gaged by γ rays. Bagasse and raw sugar are weighed during a free fall between Cs^{137} and a detector; accuracy is 1% for sugar, 5% for bagasse. Snow packs are remotely metered for water content using 40-mc Co^{60} buried in the ground, and a shielded G-M meter 15 ft above; water equivalents up to 45 in. are telemetered with 5% error (*164, 914*). S content of petroleums is detected within 0.05% in 5 min, using gage based on resonant absorption by S of Mn X-rays; Fe^{55} (5 μc) is source of these (*164*).

Backscattering is used for sorting coal from shale (NU, Oct. '54, p. 72). Soil-density measurement instruments were developed so that a ratio of 70 for backscatter to background could be reached (*164, 914*).

A 2-kg backscatter gage measures steel thickness up to 2 cm within 4%,

under wraps...

Like the tower shown here (about to leave our plant for an important A.E.C. installation) much of the nuclear component work we have done in recent years has been under the "wraps" of security. However—thanks to progress in the peaceful applications of atomic energy—it is now possible to reveal that we have been participants in this vital industry from the start. Fabricating components of complex design and metallurgical composition for A.E.C. installations at ... *OAK RIDGE ... SAVANNAH RIVER ... RICHLAND ...*

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The new Ekco *Type N550 Scintillation Counter*, shown alongside the Scaler, measures radioactive materials, in liquid or powder form, in the milli-microcurie range. For this purpose, it offers about twenty times the performance of a Geiger-Muller Counter, giving approximately three times the background count from 1 milli-microcurie of Radio-iodine. Used in leading hospitals, this equipment ensures both accuracy and speed with a minimum of skill. Type N550 operates with any Scaler which incorporates a pulse-height discriminator and provides H.T., L.T. and H.V. supplies.

Write for the complete catalogue of Ekco Nucleonic and Electronic Equipment, including Scintillation Counters, Scaling Units, Counting Ratemeters, Radiation Monitors, Vibrating Reed Electrometers, Geiger-Muller Tubes, Lead Shielding, Linear Amplifiers, Power Units, Complete Counting Installations, Thickness Gauges.

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PADDLE-WHEEL CORROSION from cavitation can be determined by painting turbine blades with radioactive paint and measuring paint wear over prescribed operational periods. Previous tests showed that paint had suitable wear during operation. Labeling was As^{76} (882)

with calibration almost independent of tube diameter. Instrument uses nearly 180-deg scattering angle, and a pulse-height analyzer set for 0.2-Mev backscattered radiation. Co^{60} source (20 μc) is mounted 2.5 cm from NaI scintillator (*463*).

Neutron gaging. Soil moisture metering relies on scattering and moderation by H. A Po-Be source attached to a BF_3 slow n counter and ratemeter is laid on the surface. Response is practically independent of presence of elements with $Z > 5$. Calibration shows H content accuracy of 0.5%. Measuring moisture in grain, timber, and stone with portable battery-operated gage is being considered (*164, 463*).

Scattered neutrons activate a Rh plate in measuring oil saturation in cores of sand packs. Fast n source above plate and core below allows local saturation measurements without interrupting the flow. Rh decays with 44 sec and 4.2 min half-life (*164*).

Brine detection in bore holes. Different layers of oil and brine can be detected through steel and concrete by measuring the prompt gammas emitted on absorption of slow neutrons. A Po-Be neutron source and γ-scintillation counter (with a one-channel kick-sorter) can detect the rise of brine in a steel pipe 1-cm thick and set in con-

Improving Beta Gages

Using a beehive-type scattered-radiation eliminator makes it possible to mount source and detector on two parallel I-beams; rather than the conventional long U-frame, when scanning along the width of a sheet is desired. With a center-axis deviation of 6 mm and a pass-level deviation of 10 mm, instrument error was <0.5% (1054).

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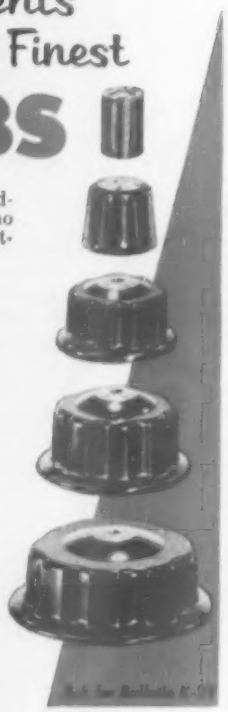

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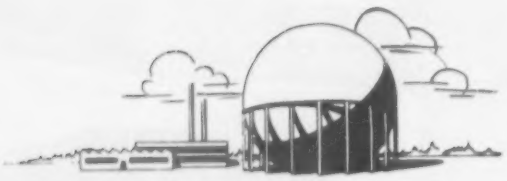
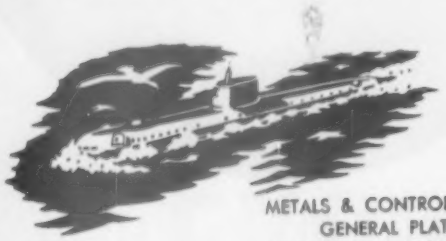


Fuel Element Materials fabricated by the General Plate Division now being used in the production of power from the atom.



METALS & CONTROLS CORPORATION
GENERAL PLATE DIVISION
ATTLEBORO, MASSACHUSETTS, U.S.A.

crete. Protecting bore holes from fouling by brine may save more money than construction of several atomic reactors (*395*).

Static eliminators. Beta-emitting static eliminators speed production. Removing the static charge lowers the fire risk considerably so that machines can be run faster. Production increases of 20–50% have been claimed in some cases (*395*).

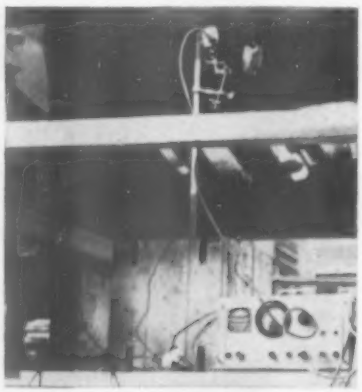
Beta-gage improvement. A surface area only 9-mm diameter is inspected by an experimental β-backscatter gage, developed to measure coating thickness on curved surfaces. Source is a capsule of Kr^{85} mounted in a 6-mm diameter collimating tube. Tube is inside a tapered light guide. A thin flat ring of NaI(Tl) is connected by the light guide to a standard photomultiplier (*463*).

Gamma-gage improvement. Because scintillation counters are much more sensitive they can compete with ionization chambers as detectors for γ-gages. Tests show that by using only a few stages of electron multiplication and a vibrating-reed electrometer, stability comparable to that of a good ionization chamber can be achieved (*463*).

Medical Applications

The major medical uses of radioisotopes have been in research (*309*). Diagnostic uses, inadequately explored, are promising. In therapy, radioisotopes have proved disappointing in their narrow field of application, but evaluation is not complete.

Research. Metabolic studies, in-



MATERIAL FLOW in Portland-cement kiln is traced by scintillation counter. Feed of limestone and clay is activated in JEEP, the 0.2% Na forming most activity. By adding about 130 gm of activated feed during operation, flow rate in different sections is found (882)

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602	30 to 400	100-300-1000	10 to 9990	x-ray therapy
603	400 to 1300	5000-15,000-50,000	500 to 499,500	On telecurie gamma sources
604	10 to 30	100-300-1000	10 to 9990	Low energy therapy
605	400 to 1300	1-3-10	0.1 to 99.9	Gamma ray sources
606	400 to 1300	10-30-100	1 to 999	Gamma ray sources
607	400 to 1300	100-300-1000	10 to 9990	Gamma ray sources
608	30 to 400	1-3-10-r/hr.	1 to 999 / 600	Survey in vicinity of x-ray machines.
612	6 to 35	100-300-1000	10 to 9990	Low energy therapy
613	30 to 400	0.1 - 0.3 - 1	0.01 to 9.9	Survey in vicinity of x-ray machines.

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dicating a much greater lability of body constituents than expected, have profoundly altered some concepts. For example, the discovery of the rapidity of exchange of mineral salts to bone and in teeth should lead to clearer understanding of their normal state and metabolic faults. The combination of the isotope method with classical enzyme isolation promises to extend the knowable number of cell reactions beyond degradation reactions.

Among the advantages of radioisotopes in research are that they permit higher accuracy than conventional methods and they permit measuring at a distance (*397*). Applications of the first advantage, together with the fact that radioisotopes can take part in living processes are: establishing, in a particular instance, that exchange of an amino acid did not occur between injected protein and free amino acids; measuring, by an isotope-dilution-type method, the amount of antibody or other precipitable protein in ground-up tissues where they cannot be extracted; estimating body spaces (e.g., plasma, lymph, cerebrospinal fluid spaces); and measuring rates of movement of body solids, gases, and fluids.

Measuring at a distance depends principally on gamma radiation. Some uses are locating brain tumors (I^{131}-diiodofluorescein or I^{131}-albumin), measuring the adequacy of new circulation in pedicle skin grafts (Na^{22}), diagnosing placenta praevia (I^{131}-albumin), estimating the vigour of blood regeneration in bone marrow (Fe^{59}), and locating retrosternal thyroid tissue and functional thyroid metastases. Development of radiographic and scanning techniques would be more convenient than collecting excreta.

Diagnosis. A number of diagnostic uses have become well-standardized procedures (*309, 1039, 914*). Among these are thyroid uptake of I^{131}, and brain-tumor localization with tetraiodophenolphthalein and P^{32}. Isotopic methods of determining blood volume, circulation time, and other aspects of cardiac function are not yet established for routine use.

Therapy. Therapeutic uses of radioisotopes have been as external or internal radiation sources (*309*).

Among the external sources are cobalt-60 (convenient substitute for radium, available in large quantities, high specific activity), cesium-137 and tantalum-182 (softer γ-rays than cobalt, competitive position in relation

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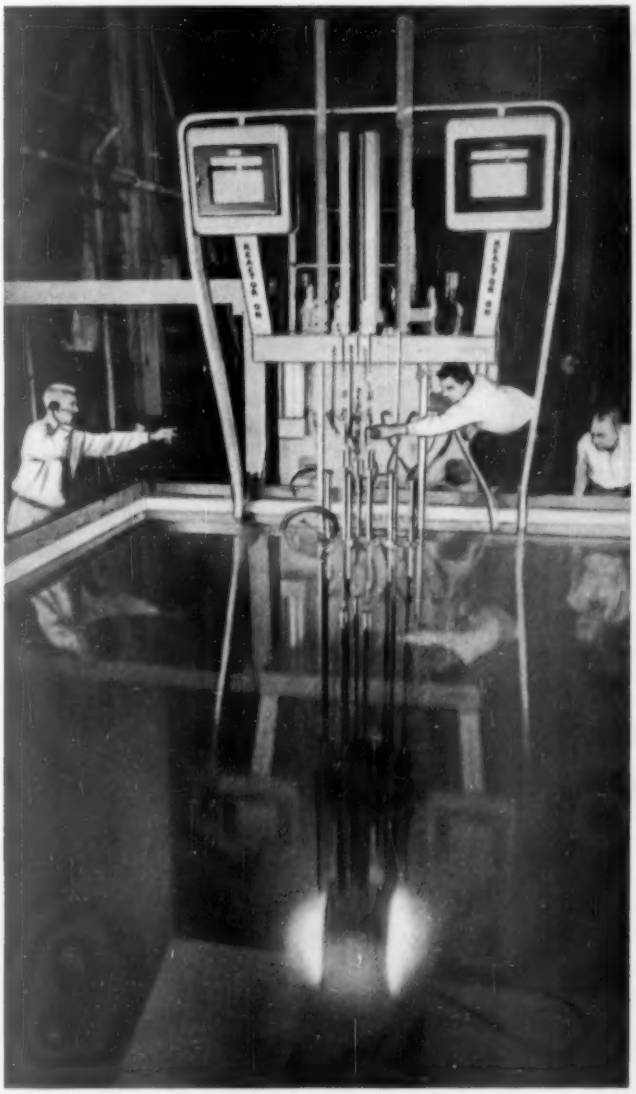

United States Exhibit Reactor at the Geneva "Atoms for Peace" Conference was designed, built, and is being demonstrated by Oak Ridge National Laboratory, operated by Union Carbide and Carbon Corporation for United States Atomic Energy Commission.

Here is an outstanding example of the peaceful applications of atomic energy. It is the operating research reactor in the United States Exhibit at the United Nations International Conference on Peaceful Uses of Atomic Energy at Geneva.

This pool-type reactor was designed and built at Oak Ridge National Laboratory, which Union Carbide operates for the United States Atomic Energy Commission. This Laboratory, in addition to its extensive research work, is also the Nation's chief source of radioactive and stable isotopes.

Union Carbide also operates the huge gaseous diffusion plants at Oak Ridge, Tennessee, and Paducah, Kentucky for the Atomic Energy Commission. In addition, the Corporation is a prime supplier of uranium and uranium concentrates from its mines and mills in western Colorado, and is a leading producer of super-high-purity graphite and carbon for nuclear applications.

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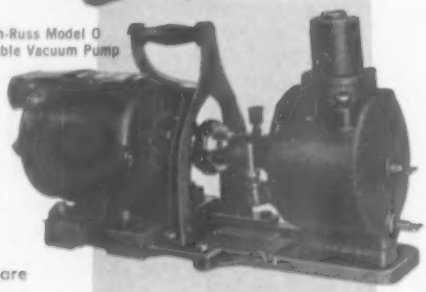

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to X-rays not established), strontium-90 (fairly intense beta emission for superficial therapy), and thalium (X-ray generator).

Some internal and external therapeutic uses have been (*914, 1039*) iodine (treatment of thyreotoxicoses, heart diseases, and carcinomas of the thyroidea); cobalt (tumors); phosphorous (blood disorders, such as leukemia and polycythemia vera, and lesions); gold (leukemia, liver metastases, and malignant effusions); and thorium-X (morbus).

Sterilizing medical supplies. Bulk medical supplies (dressings, sutures, instruments, etc.) and special items such as human bone-bank bone can be mass sterilized as soon as proper irradiation facilities are ready to operate. Lyophilized human bone, sterilized by γ-irradiation, has been highly successful on dogs, and over 30 humans have undergone successful grafts. Laboratory cobalt-60 sources will process bone, aorta, and similar material for surgical use.

Cost estimates indicate that the radiation sterilization chamber can be constructed for $57,000 and can operate at about $55,000 annually, depending on source costs. A preliminary design uses $4 \times 4 \times 4$-ft pallets and cooling reactor fuel elements, giving an estimated capacity of 120 pallets/8-hr day. This is enough to handle the production in a large manufacturing plant, and the operating cost estimate compares favorably with steam sterilization (*175*).

Benzene hexachloride production. Radiation chlorination of benzene takes place rapidly enough to make it commercially attractive. The reaction is presently conducted on a commercial scale using ultraviolet radiation.

Although the trial runs used cobalt-60, cost estimates have been made for fission-product sources to produce 454 kg/day of the gamma isomer of hexachloride. The cost, using 0.5-, 1-, and 2-year-old gross-fission-product sources, is about $1.51/kg gamma isomer in 12% mixture. The commercial cost is now $1.90–3.00. The gamma radiation process can evidently compete with the ultraviolet process; other than the chemical reactors and radiation sources, equipment is the same (*168*).

Higher doses for foods. Doses of 800,000 rep cause no detectable side reactions in meats if free-radical acceptors are used. Shelf life is increased to 3 months at 36° F (*172*).

MISSILE SYSTEMS

Research and Development

Broad interests and exceptional abilities are required of scientists participating in the technology of guided missiles. Physicists and engineers at Lockheed Missile Systems Division are pursuing advanced work in virtually every scientific field.

• • •

Below: Missile Systems scientists and engineers discuss future scientific exploration on an advanced systems concept with Vice President and General Manager Elwood R. Quesada. From left to right: Dr. Eric Durand, nuclear physicist, systems research laboratory; Ralph H. Miner (standing), staff division engineer; Dr. Montgomery H. Johnson, director, nuclear research laboratory; Elwood R. Quesada; Dr. Louis N. Ridenour (standing), director, program development; Willis M. Hawkins (standing), chief engineer; Dr. Joseph V. Charyk (standing), director, physics and chemistry research laboratory; Dr. Ernst H. Krause, director, research laboratories.

• • •

Scientific advances are creating new areas of interest for those capable of significant contribution to the technology of guided missiles.

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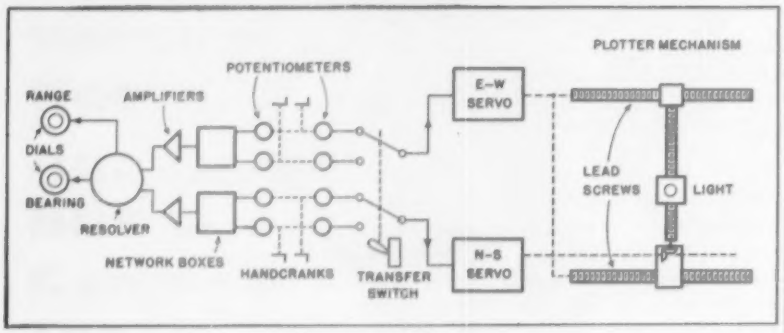

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Problem: Determine automatically and continuously the distance and relative bearing of any two points on a map.

Ford Instrument's Solution: A combination of two standard components – a map plotter and an electrical resolver.

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Here's how it works: The plotter proper has a smooth unobstructed glass top on which the map is placed. Under the glass there is a light traveling on screws. The screws are driven horizontally (E-W) and vertically (N-S) by servo motors actuated by a computing mechanism. The position of the light on the plotter is controlled by four handcranks. Two of the handcranks are used to position the light under the first point; a transfer switch is then thrown and the other two handcranks used to position the light under the second point. The map coordinates of the two points are algebraically added in two network boxes, the resultant voltage from the network boxes being the N-S and E-W distances between the two points. The resolver converts these two distances into range and bearing, which are indicated on two dials. Such a technique results in astounding accuracy. In a computer employing this principle, the maximum range error is on the order of one yard in a thousand, and the maximum bearing error is 10′. The average errors are about half the maximum.

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BIBLIOGRAPHY

These are the papers chosen for coverage in this NUCLEONICS' special report on the Geneva Conference. The numbers are those assigned by UN and bear no relation to those listed on p. 76 of the June '55 NUCLEONICS. UN advises that the conference record will be available about January 1956.

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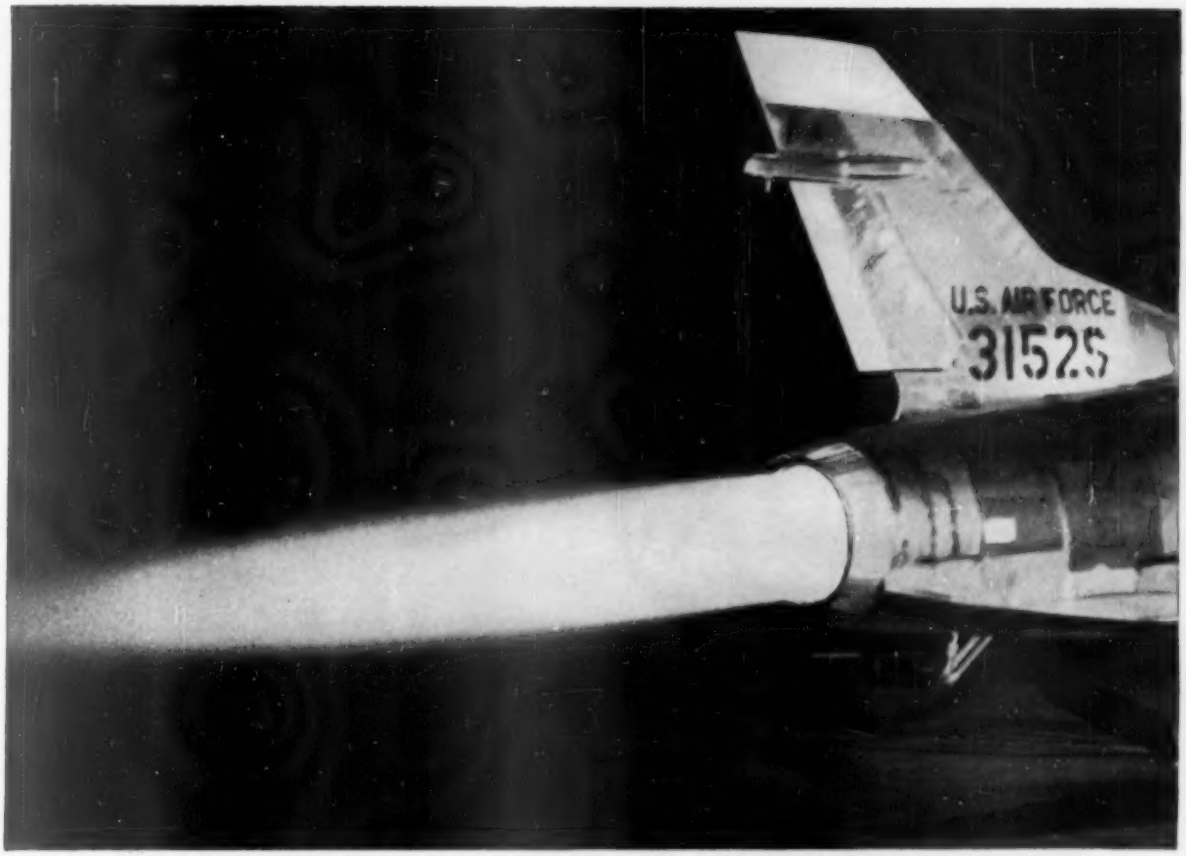

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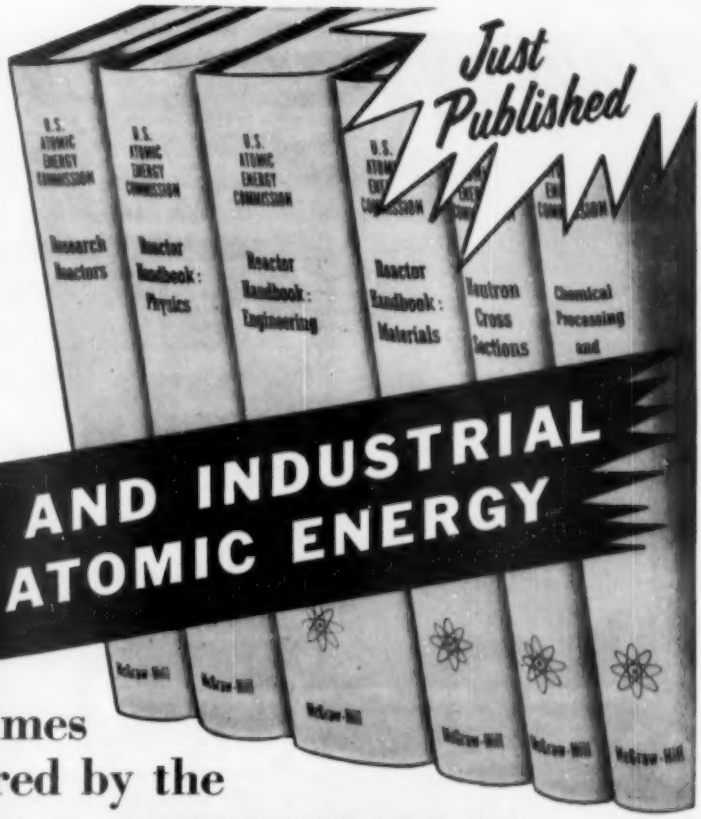

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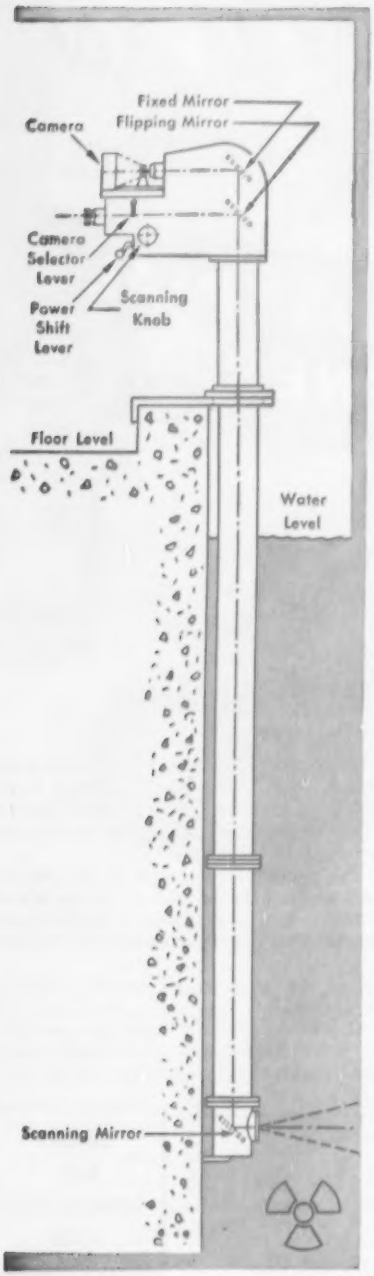

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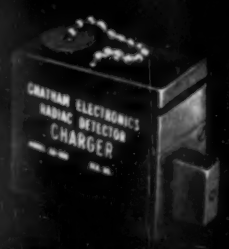

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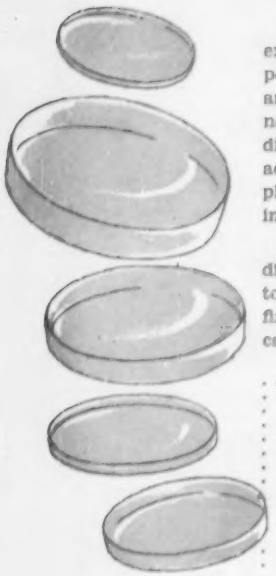

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920. Calculations for heavy water and natural uranium 10-Mw reactor, C. Salvetti, S. Gallone (Italy)
930. A new development of Geiger-tube design, J. Hermsen, K. van Duuren (Netherlands)
936. A "wet" suspension of uranium oxide for a homogeneous power reactor, H. de Bruyn, *et al.* (Netherlands)
938. A "dry" suspension of uranium oxide for a heterogeneous power reactor, H. de Bruyn, *et al.* (Netherlands)
946. Report on research reactors, L. Kowarski (UN)
947. On the possibility to improve the heat transfer of uranium and aluminium surfaces in contact, F. Boeschoten (Netherlands)
953. MTR-type fuel elements, E. J. Boyle, J. E. Cunningham (USA)
962. Energy and fuel requirements and possibilities of the FPR Yugoslavia, I. Kovacevic, B. Vukanovic (Yugoslavia)
985. The use of radioisotopes in Australia, C. E. Eddy (Australia)
1053. The field experiment of littoral drift using radioactive glass sand, S. Inoue, M. Kato (Japan)
1054. Brief review of applications of isotopes in process and quality control, M. Kato (Japan)
1061. Applications of radioisotopes to research and industrial problems in Japan, K. Kimura (Japan)
1082. Note on supply and demand for electric power in New Zealand (New Zealand)
1113. Energy requirements of Pakistan for the next twenty years and the need for nuclear power (Pakistan)
1115. Future Korean requirements for heat and power, Chullchai Park (Republic of Korea)

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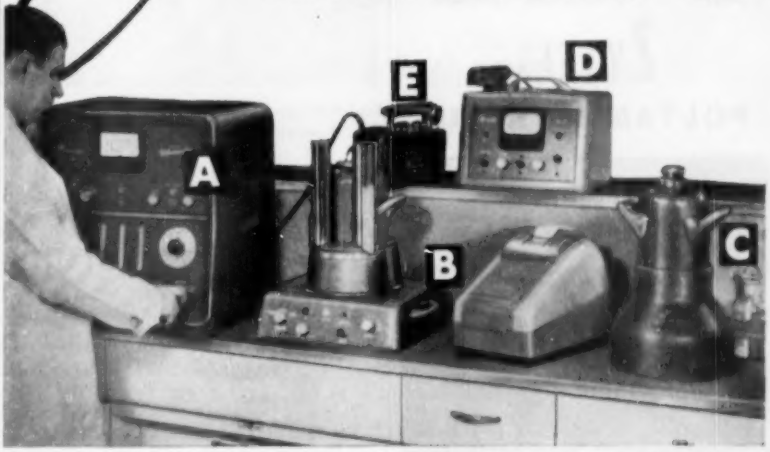

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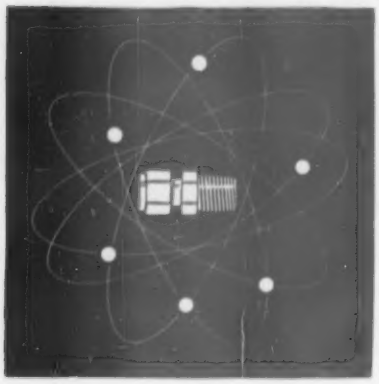

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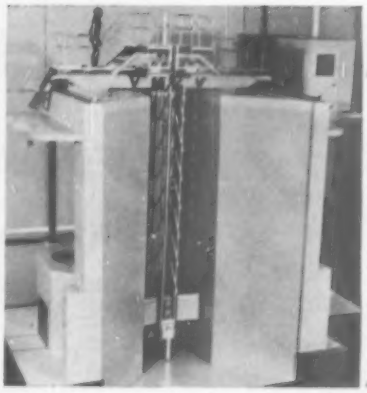
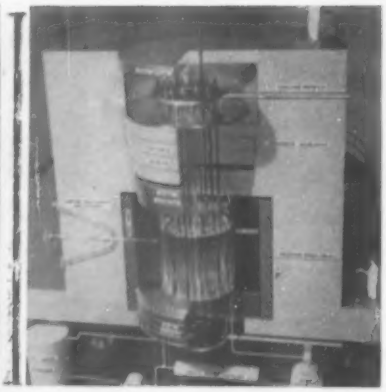


Pool-type research reactor model. American Machine & Foundry Co.



NRX reactor model. Atomic Energy of Canada, Ltd.

Preview of Forum Trade Fair

Sixty-eight American industrial organizations and a group of U. S. and foreign governmental organizations will exhibit at the Atomic Industrial Forum's Trade Fair of the atomic industry—Sheraton-Park Hotel, Washington, D. C., September 26–30.

The Trade Fair is being held in conjunction with the Forum's fall meeting and concurrent sessions of the American Nuclear Society. It is planned to bring together the buyer and seller of every essential product and service in the field.

The wide audience will include government officials and the staffs of government agencies; key management officials; members of the U. S. Congress; technical, commercial, and scientific personnel of various foreign missions in the Capital; representatives of national associations; and delegations from engineering colleges in the Washington area.

The Forum is also inviting to the exhibit all persons interested in the peacetime development of atomic energy. Tickets for admission can be obtained from the Atomic Industrial Forum, 260 Madison Ave., New York 16, N. Y. Attendance at meetings is limited to registrants on a fee basis.

In addition to industry exhibits, five displays will be entered by government organizations and research contractors—Argonne National Laboratory, Atomic Energy of Canada, Ltd., British Atomic Energy Authority (Harwell), Brookhaven National Laboratory, and Food and Container Institute of U. S. Quartermaster Corps. Their displays will cover reactor technology, metallurgical techniques, isotope applications, food sterilization, and medical therapy.

In accepting space applications, the Fair management has discouraged the display of borderline products, equipment, and services. Briefly, the products or services to be featured in each firm's exhibit are:

ALCO Products, Inc., Schenectady, N. Y. Reactor vessels, steam generators, heat exchangers, feedwater heaters, evaporators, condensers, pressure vessels, prefabricated pipe, water and process pipe, nickel plating service, and engineering services available for design and production of power-producing reactors and affiliated equipment.

Allied Chemical & Dye Corp., General Chemical Div., New York, N. Y. Process and laboratory chemicals for atomic production and research operations, including radiochemicals and reagents.

Allis-Chalmers Manufacturing Co., Milwaukee, Wis. Model of nuclear power plant for Argonne National Laboratory, showing company's roll in design and construction of power generation, heat transfer, and other special equipment for the plant.

American Machine & Foundry Co., New York, N. Y. Model of swimming-pool research reactor and control-rod-drive mechanism.

Anton Electronic Laboratories, Brooklyn, N. Y. Stainless steel halogen-quenched G-M counter tubes, corona-discharge voltage-regulator tubes, nuclear instrumentation and equipment packages for monitoring

radiation, portable radiological monitoring survey meter instruments, scalers, counting-rate meters, food comparators, beta thickness gages.

Atomic Development Mutual Fund, Washington, D. C. Information on the mutual fund's investment portfolio, reflecting investments in a variety of companies engaged in atomic activity, including uranium mining and processing, reactor component and radiation instrument manufacturing, radioisotope processing, mining and processing of special nuclear materials, and nuclear engineering.

Atomic Instrument Co., Cambridge, Mass. Instrumentation for detection, measurement, and analysis of radioactivity, including single- and multichannel differential pulse-height analyzers, scintillation spectrometers, special scintillation spectrometers, special scintillation counting assemblies, scalers, linear amplifiers, high-voltage supplies, and electronic counting and control instruments.

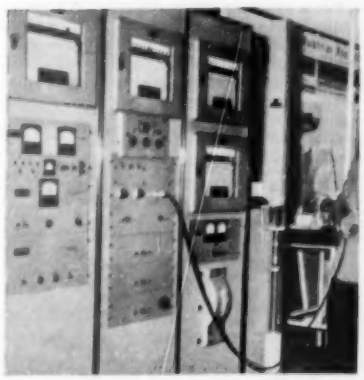
Babcock & Wilcox Co., New York, N. Y. Nuclear steam-generation plants and components.

Battelle Memorial Institute, Columbus, Ohio. Applications and utilization of radioisotopes and the development of reactor materials and components.

Bengston & Co., New York, N. Y. Information showing range and scope of investment opportunities available in nuclear industries, with emphasis on investment companies with shares representing selection and diversification of investments in nuclear fields.

S. Blickman, Inc., Weehawken, N. J. Enclosures for handling radioactive materials.

Brush Beryllium Co., Cleveland, Ohio. Fabricated beryllium shapes



Reactor control and monitoring instruments. Minneapolis-Honeywell Regulator Co.

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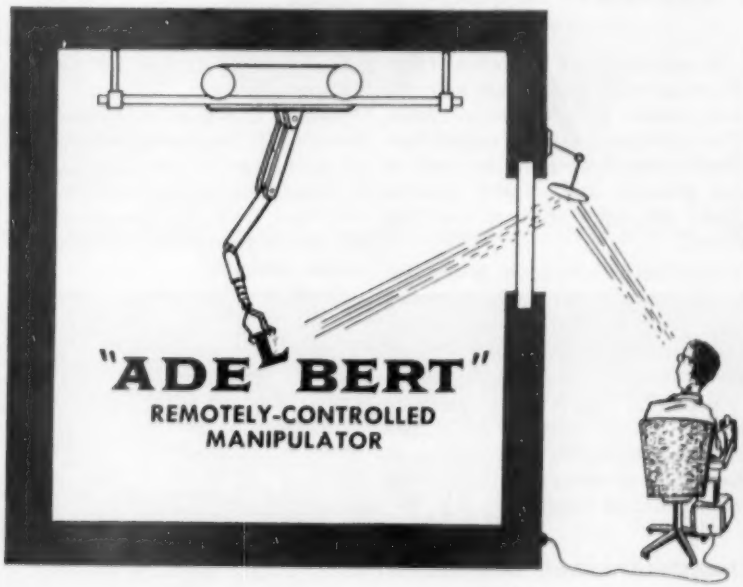

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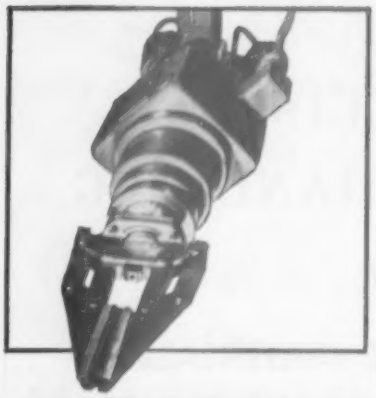

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Byron Jackson Co., Los Angeles, Calif. Pumps and mechanical seals engineered for nuclear-power application; electronic instruments for control of nuclear processes.

Carborundum Metals Co., Niagara Falls, N. Y. Applications of use of zirconium in nuclear power plants.

Catalytic Construction Co., Philadelphia, Pa. Model of uranium ore refinery and outline of services in project design, process design, engineering and construction of plants to process uranium ore and produce uranium metal, chemical recovery plants, laboratories, and pilot plants.

Central Research Laboratories, Red Wing, Minn. Master-slave manipulators, plus description of design and manufacturing services for manipulator accessories.

Central Scientific Co., Chicago, Ill. Beta-ray H/C meter for measurement of percentage of hydrogen in liquid hydrocarbons, utilizing absorption of beta rays from strontium-90 source, for service in research, product, and routine analysis.

Chatham Electronics, Livingston, N. J. Scintillation counters, rate meters, dosimeters, and G-M counters.

Chempump Corp., Philadelphia, Pa. Leakproof fluid transfer equipment, including a seal-less centrifugal pump, axial flow units, and high-pressure units.

Combustion Engineering, Inc., New York, N. Y. Complete nuclear steam-generation plants, including development, design, and manufacturing of reactor vessels, steam generators, and other components.

Consolidated Edison Company of New York, New York, N. Y. Representation of projected nuclear power plant to be built at Buchanan, N. Y.

Corning Glass Works, Corning, N. Y. Glass for radiation shielding windows, and dosimeters.

Crane Co., Chicago, Ill. Specialized valves, fittings, and fabricated piping developed for nuclear installations.

Daystrom Instrument, Archbald, Pa.; **Weston Electric Instrument Corp.,** Newark, N. J. Nuclear reactor-control equipment and instrumentation, including analyzers, amplifiers, recorders, scalers, thermometers, radiation densitometers, relays, product resolvers, integrators, fluxmeters, and tachometers.

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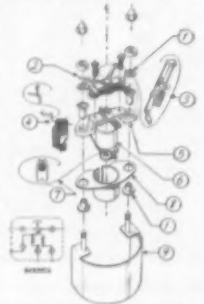
Assembly Products, Inc., Chesterland 28, Ohio.



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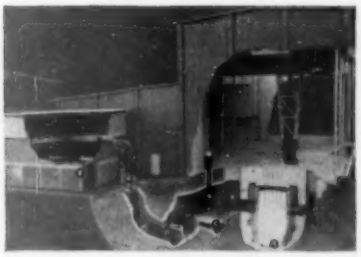
Booth A-150, Instrument Show, Sept. 12–16, L. A.



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Foster Wheeler Corp., New York, N. Y. Models of nuclear power plant (aqueous homogeneous power breeder), research reactors, and chemical plants and components for nuclear energy applications.

General Dynamics Corp., New York, N. Y. Display based on first nuclear-powered submarine, USS *Nautilus.*

General Electric Co., Schenectady, N. Y. Power reactors, research reactors, and nuclear-power equipment components (illustrated by scale and cutaway models with animation).

General Mills, Inc., Minneapolis, Minn. General Mills mechanical arm.

Griscom-Russell Co., Massilon, Ohio. Heat transfer and control apparatus, steam generators and superheaters, gas turbine air heaters, and regenerators.

High-Voltage Engineering Corp., Cambridge, Mass. Van de Graaf particle accelerators used in X-ray therapy, radiation sterilization and processing, radiography, and research in chemical, nuclear, and biological fields.

Holmes & Narver, Inc., Los Angeles, Calif. Models and technical reports on shielding and blast effects.

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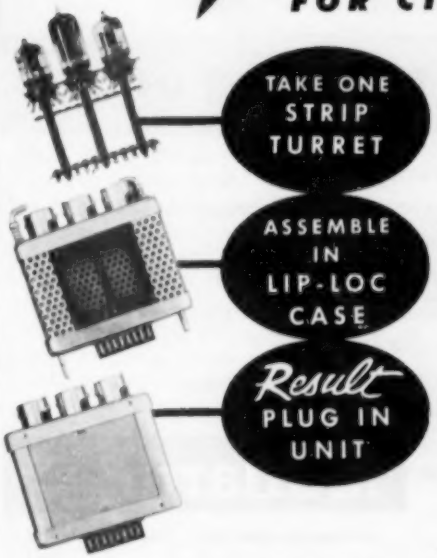

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Sr^{90}	25 years	3/8 inch	75	300	400	125	50
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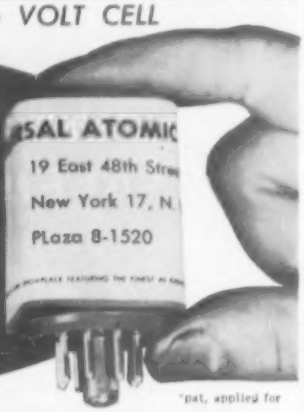

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Newport News Shipbuilding & Drydock Co., Newport News, Va. Model of Mariner-class vessel showing possible use of nuclear power for propulsion, and model of 5-Mw homogeneous reactor showing welding and fabricating techniques developed for construction of zirconium-alloy core tank and stainless-clad pressure vessel.

North American Aviation, Inc., Los Angeles, Calif. Power reactor and medical reactor models.

Norton Co., Worcester, Mass. Fusion-stabilized high-purity minerals

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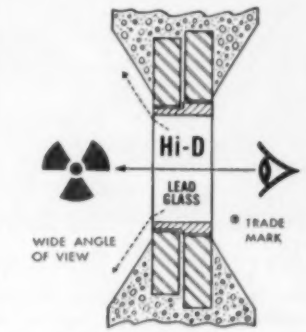

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Nuclear Science and Engineering Corp., Pittsburgh, Pa. Low-level anticoincidence apparatus.

Nucleonics Magazine, New York, N. Y.

Panellit, Inc.; Jordan Electronics, Inc., Alhambra, Calif. Radiation detection and monitoring equipment, alarm systems.

Radiation Applications, Inc., New York, N. Y. Applications of nuclear radiation to commercial products and processes, consultation and research services.

Republic Steel Corp., Cleveland, Ohio. Titanium and stainless steel and other basic metals and alloys.

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Sylvania Electric Products, Inc., Bayside, N. Y. Fuel-element shapes typical of wide cross section of fuels being produced or under development for nuclear reactors, description of production and development facilities, and model of nuclear reactor.

Technical Measurement Corp., New Haven, Conn. Radiation instruments, scalers, rate meters, high-voltage power supplies, linear amplifiers, liquid scintillation counters, analyzers, and custom instruments to be used for

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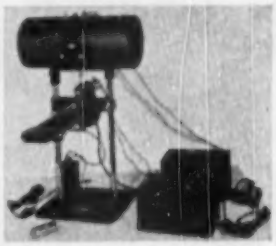

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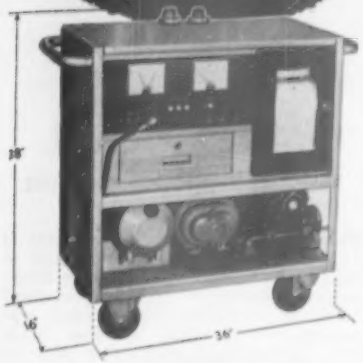

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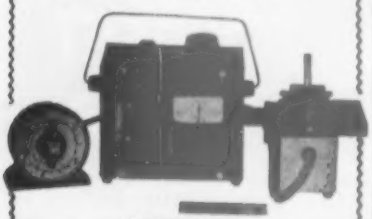

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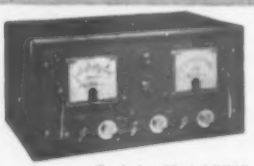

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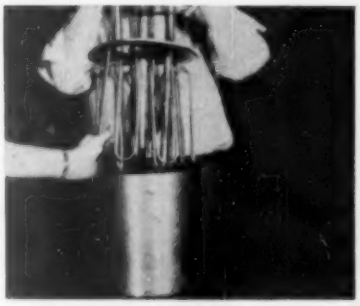

FUEL CYLINDER of Raleigh Research Reactor developed leak detected in early June. Fuel solution was drained, cylinder was removed and shipped to ORNL for failure analysis. Picture shows cylinder and coolant tubes before reactor assembly. Babcock & Wilcox Co. is making a replacement unit. Operation is due to resume in September or October. Meanwhile, opened reactor assembly is being modified to minor extent; main addition is a food and plastics irradiation cell that will get high-intensity gammas from gaseous fission products circulated from core

NRX contaminated, shut down. Special fuel rod being tested in Canada's NRX reactor has contaminated the moderator and forced a several-week shut down for D_2O repurification and system decontamination. Shutdown and D_2O dump were automatic—only minor amount of activity got into cooling water going to delay tanks. Main operating room was not contaminated; situation does not compare with previous breakdown (NU, Jan. '53, p. 76).

Convair to build medical reactor; sets up separate A-plane group. General Dynamics' Convair Division at Fort Worth, Texas, plans to build a 1-Mw fuel-plate-type reactor at Albuquerque, N. M., for the Lovelace Foundation for Medical Education and Research. To be completed before '57, the complete facility will cost $2½-million.

R. N. Little, chief nuclear physicist at Convair-Fort Worth, is acting head of a new department set up for continued research and development work on nuclear-powered aircraft.

Chicago and Detroit reactor plans OK'd by AEC. By giving "promise of significantly advancing power reactor technology and providing an acceptable basis for negotiation under the Power Demonstration Reactor

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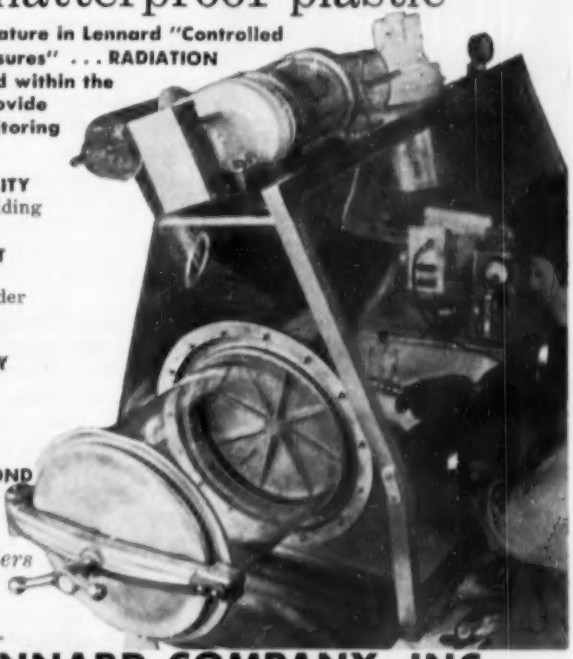

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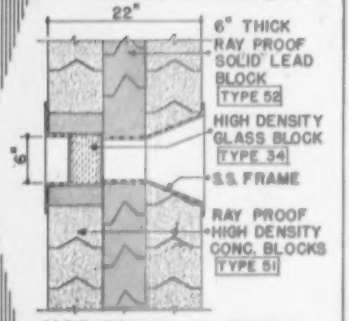

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program," reactor projects proposed by Detroit Edison Co. and Associates and by Nuclear Power Group led by Commonwealth Edison Co. of Chicago have gained AEC's approval for further negotiation (NU, May '55, p. 11). While turning down proposals from Yankee Atomic Electric Co. and from Consumers Public Power District of Columbus, Neb., the Commission left the door open for further discussion. Although no reasons were given for the rejections, it is understood that all original proposals fell short of AEC's aims. Reportedly, the two successful firms lessened their provisions for AEC financial and insurance aid.

Combustion Engineering to be third Navy reactor supplier. Under a contract in which the firm will provide plant facilities, capital equipment, and cover 25% of research and development costs up to $3-million, Combustion Engineering Inc., has been selected by AEC to build and test a Submarine Reactor Small (SRS) for Navy use (NU, April '55, p. 11).

AEC funds to build ETR. Construction of the Kaiser-designed Engineering Test Reactor will be started this winter in Idaho (NU, Aug. '55, p. 10; April '55, p. 11). Construction contractor has not yet been selected. Start-up is scheduled for spring, 1957. Estimated cost is $15,000,000.

AEC requests bids for designing, fabricating core-testing reactor. To be built at the National Reactor Testing Station, Idaho, reactor will be high-pressure-water moderated and water cooled. Reactor cores will be tested under severe conditions to determine safe operating limits. Reactor will cost $250,000–500,000; delivery will be mid-1956. Proposals can be submitted in late September, 1955.

Bell Aircraft retains Nuclear Science and Engineering as advisors. Nuclear Science Corp. will help evaluate nuclear-technique applications in Bell Aircraft Corp. operations and participation of Bell in the nuclear field.

AEC sets U and D_2O charges. For research-reactor use, AEC has set lease charges of $25/gm U^{235} in 20% enriched U and sale prices of $40/kg for natural U and $28/lb for D_2O. The D_2O price is reportedly based on production costs.

Union Carbide forms nuclear firm. Union Carbide Nuclear Co., a division

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of Union Carbide and Carbon Corp., has been set up to integrate the Corporation's activities in the nuclear field. Union Carbide processes uranium ores and operates AEC's Oak Ridge and Paducah facilities.

8 sponsors lined up for AMF-suggested research reactor. American Tobacco Co., Continental Can Co., Corning Glass Works, International Nickel Co., Chas. Pfizer & Co., Socony Mobil Oil Co., and U. S. Rubber Co. are joining American Machine & Foundry Co. to build the $1.5–2-million Industrial Reactor Laboratories on a 250-acre site in the New York area (NU, March '55, p. 10).

Socony Mobil plans also call for a nuclear research center to start operations next year. At the center will be a Van de Graaff, and radioisotope-handling facilities.

Government to insure nuclear installations only as last resort. Clark C. Vogel, AEC deputy counsel, says AEC will enter insurance business only on the basis of demonstrated need and if lack of coverage deters participation in the atomic-energy field. Justification of need is the job of potential licensees; AEC will not attempt conclusions until all evidence is in.

Two members of an insurance counsel panel and the AEC insurance industry study team indicate that pooling of private insurance capital will be necessary. They also indicate that an adequate program, without government aid, can be worked out.

AEC issues 81 access permits in July. Total number of permits issued to organizations and individuals for access to classified information on atomic-energy technology is 229 since April of this year.

Stanford Research Institute plans 10-Mw reactor. To be built and operated by SRI near Menlo Park, Calif., reactor will be devoted exclusively to West Coast industry research. Preliminary studies to determine costs, safeguards, and prospective operating procedures have been completed.

Canada to discontinue special-price uranium-purchase contracts after March 31, 1956. Until then it will continue to pay special prices to those companies now under contract with Eldorado Mining and Refining, Ltd. Also, such contracts will not be negotiated up to that date unless production by the purchaser begins before April

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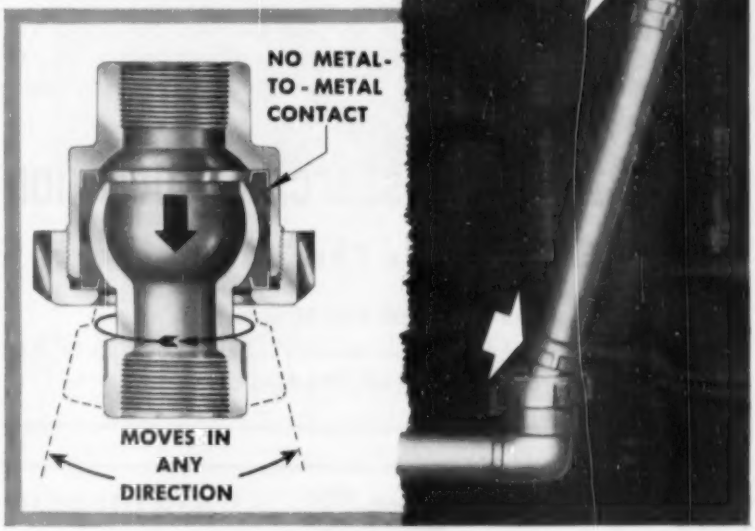

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1, 1957, and deliveries are completed by March 31, 1962. Beyond this last date, no Canadian policy has been formulated because of uncertainty regarding civilian and military requirements.

Britain builds ZEUS. An ~60-watt full-size version of the Dounreay Fast Reactor has been built at Harwell to provide design and operation data for the Dounreay reactor. ZEUS (Zero Energy Uranium System) should be operating by year end. Although much larger than the Pu-fueled ZEPHYR, which has a conversion ratio of 2, its breeding margin is less. ZEUS' enriched fuel is in long tubes mounted vertically in a hexagonal core.

Italy builds atomic research station. The National Committee for Nuclear Research has started constructing a 10-acre station at Frascati, 10 miles south of Rome; it will include one of Europe's largest synchrotrons. Operations are planned to start within 2 yr.

UK reduces radioisotope prices. Growth of demand and improved manufacturing have permitted reducing I^{131} and Au^{198} prices about 20%. Minimum charges for the isotopes, shipped from Radiochemical Centre, Amersham, are I^{131} (carrier-free solution), £1 for 1 mc; I^{131} (sterile solution), £2 for 1 mc; Au^{198} (colloidal solution), £5 for 50 mc.

Brazilian research reactor to be built under U. S. bilateral. (NU, June '55, p. 17.) Reactor, set up at Sao Paulo University, will be fueled with 6 kg of U^{235} in enriched uranium. Expense will be shared, but fuel and many of the technicians will come from the U. S. Plans are to train technicians for power plants and develop domestic ores toward self-sufficiency.

Uranium-oxide world price $10/lb. With last year's price generally at $7.25, the Australian Government recently announced a price of $10. Latter price is expected to last for about 10 yr. Four of Canada's leading five uranium mines apparently get just over $10 for average grades of 0.1% and the fifth (average 0.2%) apparently gets $8.50.

UK pushes export of nuclear plants. In promoting nuclear-plant export, bilateral agreements, similar to those of the U. S., have been signed with most Commonwealth countries. France and Denmark are two of the latest on the Continent to sing.

British Government maintains the exclusive right to manufacture fuel elements, which also must be returned to Britain for chemical processing. Security and red tape are kept to a minimum. The Atomic Energy Authority has turned over all patents for 5% royalty on sales, subject to security limitations.

Export volume may be limited by fuel supply and number of trained personnel. With the Government sole producer and operator of one batch-method plant, industry appears worried about delivery delayed by fuel availability; the problem should only be a short-term one. Limited training facilities may make difficult the expansion of the domestic program as well as providing training for importing countries.

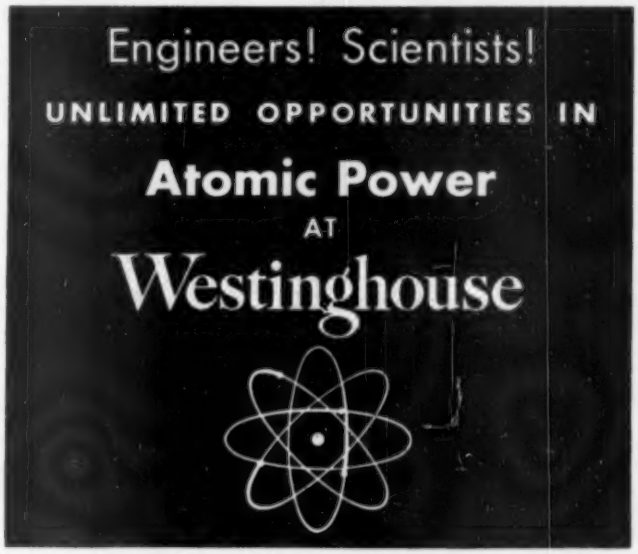
NEWSMAKERS

Col. Alfred D. Starbird, of the office of chief of engineers, U. S. Army, has been appointed director of AEC's military application division to succeed **Kenneth E. Fields,** now general manager. He will supervise the Commission's weapons research, development, testing, and production programs.

Rear Admiral H. G. Rickover, chief of AEC's naval reactors branch and director of the nuclear power division of the Navy's bureau of ships, has been awarded the George Westinghouse gold medal award by the American Society of Mechanical Engineers. The citation honors him for "eminent achievement in the power field of mechanical engineering as exemplified by the successful completion of the submarine USS Nautilus, in which his foresight, determination, and leadership, the idea of nuclear power generation was transformed from speculation to reality."

Lloyd R. Zumwalt, formerly director of the western division of Tracerlab, Inc., has been elected operations vice president of Nuclear Science and Engineering Corp.

Vice Admiral Willard A. Kitts, 3rd, USN (ret.), former manager of the industrial atomic products study within GE's atomic power equipment department, has been named acting general manager of the department. He succeeds **Francis K. McCune,** who is a General Electric vice president and general manager of the company's atomic products division.



In the suburbs of Pittsburgh, Pennsylvania, research and development is being conducted on nuclear reactors to generate power for a commercial power plant and to provide propulsion for naval vessels. In this plant the nuclear reactor for the submarine Nautilus was designed and constructed.

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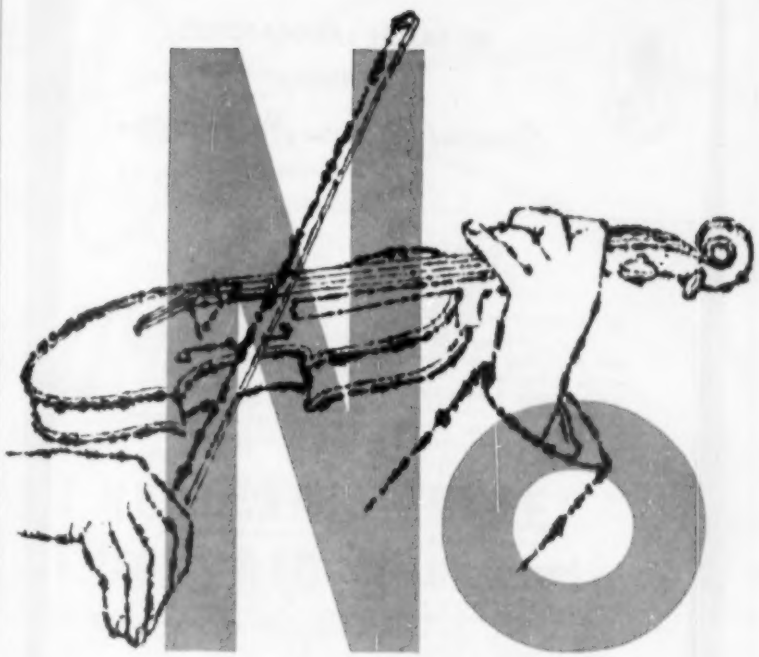

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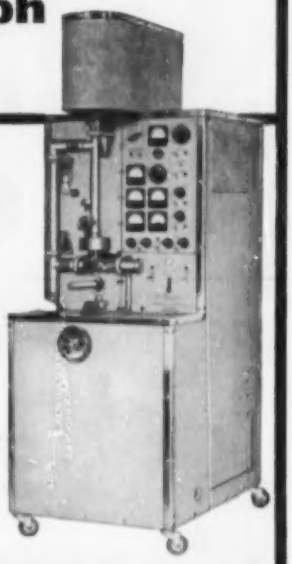

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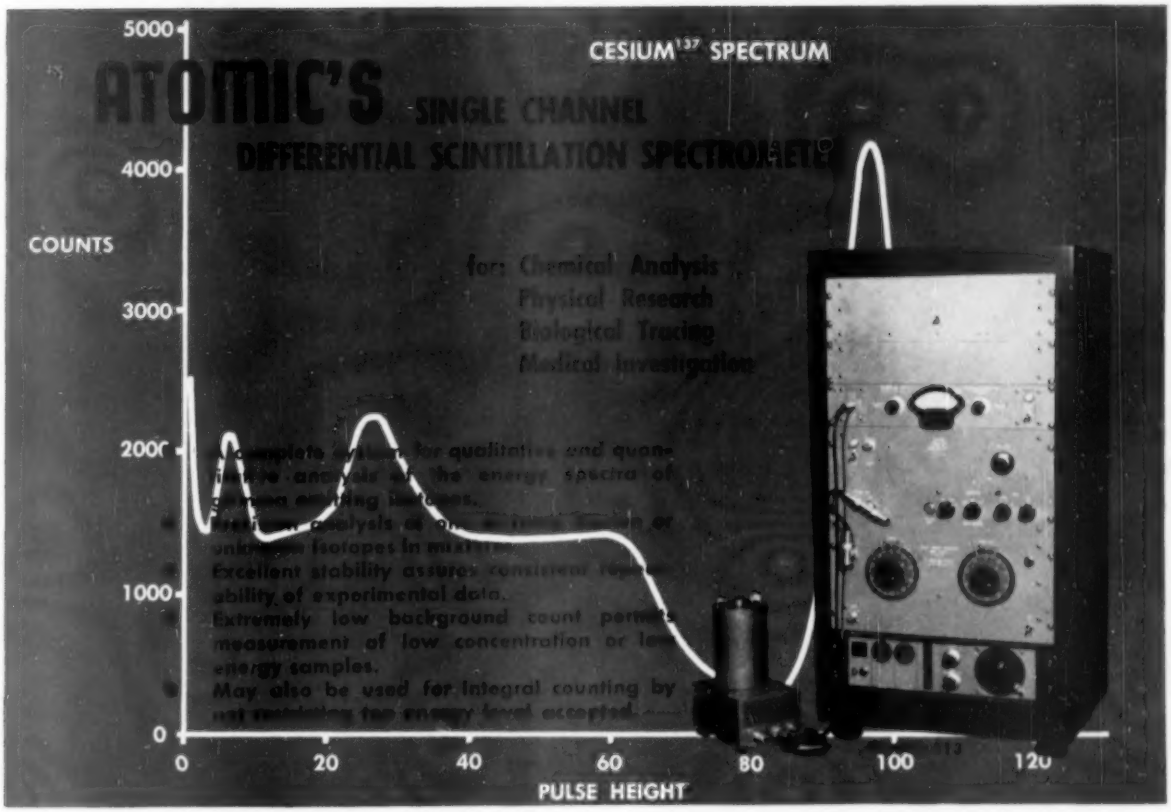

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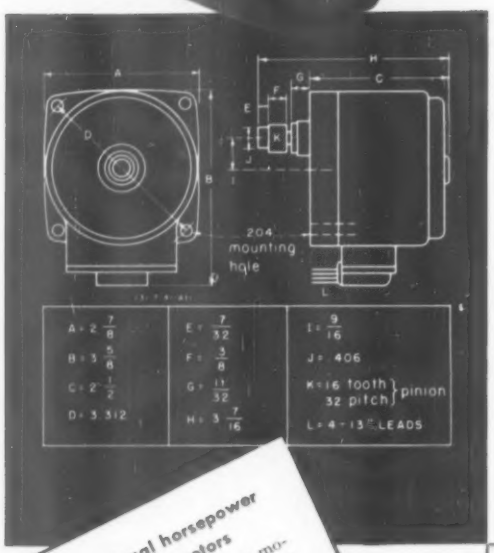

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